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INFORMAL REPORT

TMI-2 FISSION PRODUCT INVENTORY ESTIMATES (DRAFT)

E. L. Tolman
M. Nishio

EG&G Idaho

Work performed under DOE Contract No. DE-AC07-76ID01570

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ABSTRACT

This report presents the results of analyses performed through mid-FY-87 to estimate the inventory and distribution of selected radioisotopes within the TMI-2 reactor system. The intent of the report is to document the method used in estimating the fission product inventory and associated uncertainties. Since examination of the degraded core materials is not complete, the values presented should be viewed as preliminary. Selected radioisotopes for which best-estimate inventories and uncertainties are presented include: Krypton (Kr-85), Cesium (Cs-137), Iodine (I-129), Antimony (Sb-125), Ruthenium (Ru-106), Strontium (Sr-90), Cerium (Ce-144), and Europium (Eu-154).

The TMI-2 inventory data will provide a basis for relating the fission product behavior during a large-scale severe accident to smaller-scale experimental data and fission product behavior modeling work. This is an important link in addressing the many technical questions that relate to core damage progression and fission product behavior during severe accidents.

SUMMARY

TMI-2 post-accident fission product inventories and uncertainties have been estimated for selected radioisotopes using:

- the known end-state configurations of the degraded core materials and the estimated masses for each degraded core zone.
- recent laboratory examination data characterizing the fission product concentration of previously molten core material samples taken from the lower plenum region.

The information presented is an extension of previous work and includes a discussion of the methods for quantifying uncertainties in the inventory values.

Inventory values for selected radioisotopes are summarized below.

Fission Product	Estimated Inventory	
High Volatility		
Kr-85	86%	(±4%)
Cs-137	104%	(±3%)
I-129	68%	(±4%)
Medium Volatility		
Sb-125	43%	(±3%)
Sr-90	112%	(±10%)
Low Volatility		
Ru-106	48%	(±3%)
Ce-144	100%	(±8%)
Eu-154	85%	(±7%)

The estimated noble gas inventory agrees to within about 15% of the best-estimate calculated total core inventory. However, there appears to

be a wide range in accountability for all other fission product groups (i.e., high-, medium-, and low-volatility groups).

In the high-volatility fission product group, cesium appears to be accounted for to within 10%; however, up to 30% of the iodine has yet to be accounted for.

In the medium-volatility fission product group, the fraction of the core inventory of antimony accounted for is less than 50%, while the strontium inventory agrees with the best-estimate calculated total core value.

In the low-volatility fission product group, the ruthenium inventory appears to be low (less than 50%), while the inventory for cerium and europium appear to be within approximately 10% of the calculated values.

Data from the following fission product repositories were considered in estimating the post-accident inventories:

In-RCS Repositories		Ex-RCS Repositories
Degraded Core Regions	Reactor Cooling System	
Upper core debris	Hot leg piping surfaces	Reactor building water
Previously molten core material	Upper plenum surfaces	Reactor building sediment
Partially intact fuel rods	Steam generator surfaces	Reactor building lower walls
Previously molten core material in the core former zone[a]	Pressurizer surfaces	Reactor building upper surfaces
Previously molten core material in the core support assembly (CSA) region	Steam generator sediment	Reactor building air space
Lower plenum debris	Pressurizer sediment	Auxiliary building liquid
	Makeup/purification demineralizer sediment	Auxiliary building gas release
	Reactor coolant drain tank	
	RCS coolant	

a. Core former zone includes that region containing the core former plates and baffle plates.

Several major assumptions were used in estimating the above inventories. These include:

1. The specific activity (μCi/g) of those regions containing previously molten core materials for which sample examination data are not yet available (e.g., molten core zone, the core former regions at the core periphery, the core support assembly regions below the core) is assumed to be the same as the measured data from the previously molten lower plenum debris.

2. The pre-accident total core inventory for the isotopes considered is not measurable and is assumed to be given by the ORIGEN2 calculated values. No error has yet been asosociated with these calculated values.

3. No uncertainty has been associated with the "typicality" of the limited samples for each unique degraded core zone in representing the estimated total mass of degraded core material from that zone. Additional analyses will be required to estimate the variability in the measured degraded core actiities as the data become available.

4. Fission product retention in the intact rod regions was estimated based on the end-state core configuration as inferred from core drilling and visual inspection data, and from the following assumptions:

   - the retention of the higher volatility fission products within the fuel is 95 ± 5%

   - the retention of the medium and lower volatility fission products within the fuel is 99 ± 1%.

Because limited activity data are currently available from most degraded core material zones (necessitating the assumptions summarized above), the above inventory values must be viewed as preliminary and are expected to change as more core material examination data become available. The uncertainties associated with the inventory values are recognized to be relatiavely small, because data do not presently exist to resolve the above assumptions in data variability. As additional data allows resoltuion of these assumptions, the uncertainties in the inventory values are expected to increase significantly.

CONTENTS

FIGURES

TABLES

TMI-2 FISSION PRODUCT INVENTORY ESTIMATES

1. INTRODUCTION

The TMI-2 accident severely damaged the reactor core causing significant release of fission products from the fuel. Recent defueling data have confirmed that as much as 35-45% of the core material melted[1] and an estimated 15-20% of the original core material relocated to the lower plenum region of the reactor vessel in a molten state.[2] Because of this extensive core damage, the TMI-2 accident offers a unique opportunity to extend our knowledge of: important physical mechanisms affecting the initial core relocation of core material leading to a non-coolable core configuration; migration and interaction of molten fuel with core support structures, reactor vessel coolant, and the vessel lower head; and the associated fission product behavior resulting from these processes.

The TMI-2 Accident Evaluation Program[3] is being conducted for the DOE as a severe accident research effort and has the primary goal of establishing a consistent understanding of the mechanisms controlling the core damage progression <u>and</u> the resulting fission product behavior. This improved understanding is being developed through:

- examination of the end-state degraded core materials and damage to the core support structures,

- interpretation of the TMI-2 data recorded during the accident relative to the reactor system thermal hydraulic response (including core damage progression), and

- analysis work to integrate these data into a consistent scenario of core damage progression and fission product behavior.

A key part of the TMI-2 research yet to be completed is related to fission product behavior. Three major efforts are currently underway to understand the important aspects of fission product behavior resulting from the accident. These are:

- evaluate the total quantity of the measurable fission products and their distribution in the reactor systems,

- estimate the fission product behavior (release and transport) during the high-temperature portion of the accident, and

- analysis work to relate these results to core damage progression and source term technical issues.

This report addresses the first of these tasks, i.e. completing a best-estimate inventory of the measurable TMI-2 fission products.

Measurement of the fission product release from the core has been a major focus since the accident and was essential to the early estimates of core damage. It was recognized from the early radiation measurements that the release of radioactive iodine was orders of magnitude lower than expected. Extensive radionuclide measurements have been made since the accident in support of plant recovery work. The radiological inventory estimates based on these data have been published.[4]

The results documented in this report are an extension of the work presented in Ref. 4 and include an estimate of uncertainty in the calculated inventory for the measurable (longer-lived) isotopes. In addition, more recent data derived from core defueling and laboratory examinations of core material samples are included.

The report is organized as follows. Section 2 briefly summarizes the best-estimate core damage progression scenario and identifies those mechanisms affecting the fission product release and transport during the period of major core damage progression. Also included is a summary of the relative activity versus time from accident initiation for those radioisotopes considered important for severe accident source term analysis. Section 3 describes the methodology for calculating fractional core activity inventories and associated uncertainties and presents the results of such calculations using the available TMI-2 data. Section 4 presents important conclusions and recommendations for completing the fission product inventory work.

2. OVERVIEW OF TMI-2 FISSION PRODUCT RELEASE AND TRANSPORT DURING THE ACCIDENT AND SUMMARY OF MEASURABLE ISOTOPES

This section provides a synopsis of the important mechanisms affecting fission product behavior during the TMI-2 accident and a brief review of the radioisotopes considered important during severe accidents and their decay characteristics.

The best-estimate core damage progression scenario (accident scenario) provides the basis for evaluating the fission product release during the accident. The estimated fission product releases based on the best-estimate bounding core temperatures can then be compared to the end-state fission product retention measured from the various forms of degraded core material to evaluate consistency between the TMI-2 data and other severe fuel damage experiments and modeling work.

The best-estimate accident scenario[5] suggests that the major core damage progression can be divided into the following time periods after accident initiation:

1. 100-174 minutes--initial core heatup and degradation forming a non-coolable configuration

2. 174-176 minutes--B-loop pump transient

3. 176-224 minutes--degraded core heatup

4. 224-230 minutes--core failure and molten core material migration

5. 230 minutes-15.5 hours--core cooldown and reactor system cooling recovery.

Each of these periods will be briefly discussed to provide the necessary background for interpreting the inventory estimates presented in Section 3.

Initial core damage (100-174 minutes). Prior to 100 minutes, the two-phase forced convection through the reactor vessel provided the necessary cooling to prevent core heatup. At about 100 minutes, the last primary coolant pump was shut off. The liquid in the reactor vessel and upper elevations of the hot legs rapidly settled into the vessel, resulting in a vessel liquid level estimated to be near the top of the active fuel. The liquid in the reactor vessel continued to decrease as a result of primary coolant letdown and loss out the pressurizer relief valve. The top of the core started to uncover shortly after 100 minutes, resulting in heating of the upper regions of the fuel rods. By approximately 140 minutes, the core liquid level had dropped to below the mid-core level and the upper regions of the fuel had heated sufficiently to result in cladding burst (~1100 K). About this time, the operators discovered the faulty pressurizer relief valve and manually closed the pressurizer block valve, thus limiting further loss of coolant (and fission products) from the reactor coolant system.

By 150 minutes, it is estimated that peak core temperatures exceeded 1800 K and a significant amount of molten zircaloy (fuel rod cladding) and some dissolved UO_2 fuel relocated to the lower regions of the core. This relocated material solidified around the intact fuel rods, forming a consolidated region of core materials near the liquid level interface at about the 0.6-0.9 m (2-3 feet) axial elevation.[a]

From 150-174 minutes, the core heatup continued, resulting in increased core damage both radially and axially. By 174 minutes (just before the B-pump transient), the damage state of the core was extensive as shown in Fig. 1. It is estimated that peak core temperatures above 2200 K were achieved for times ranging from 20-25 minutes during this period of the accident. Fission product release under these conditions, based on the NUREG-0772 release correlations (summarized in Fig. 2), would be extensive as shown in Table 1.

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a. All references to core axial elevation in this report will be relative to the bottom of the active fuel.

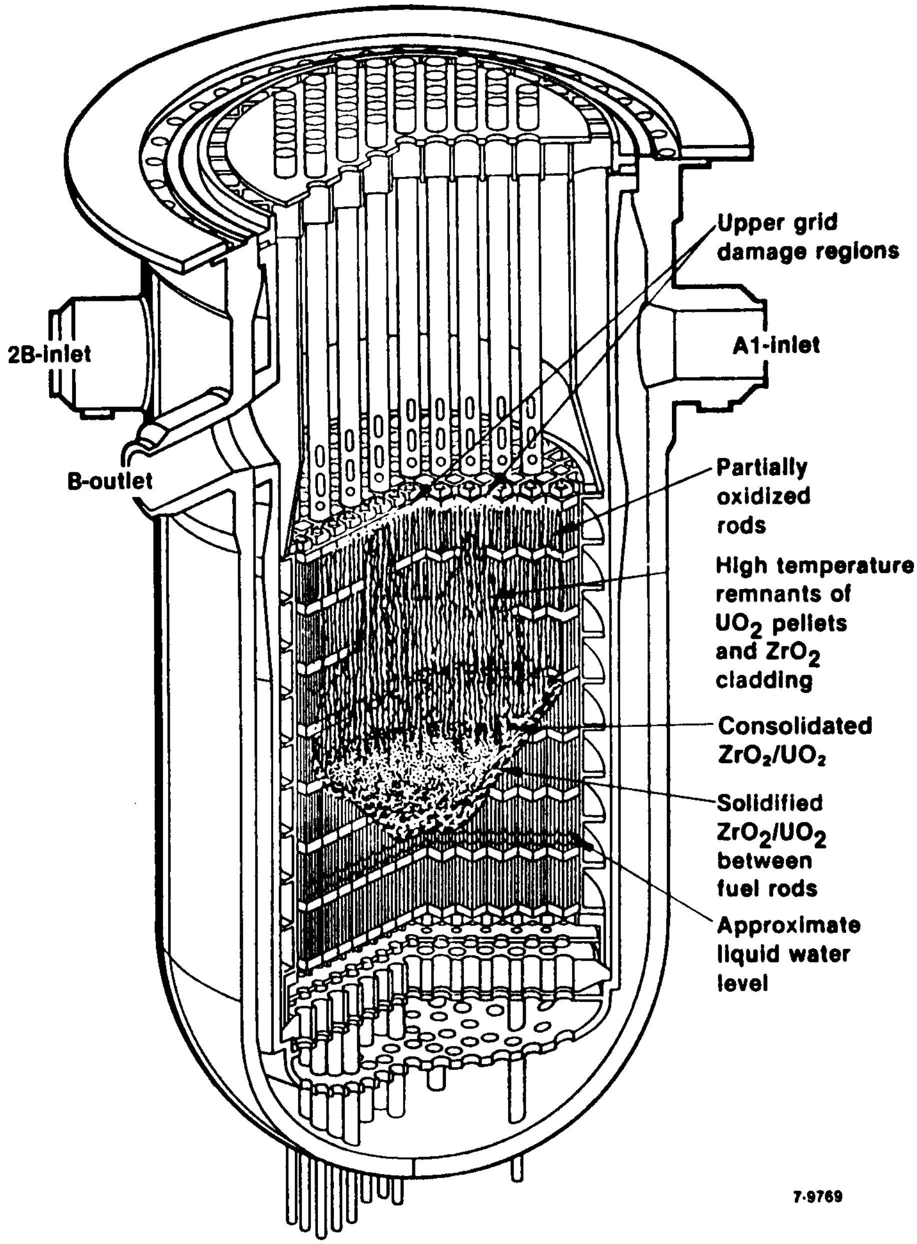

Figure 1. Hypothesized degraded TMI-2 core configuration just before the B-pump transient (174 minutes).

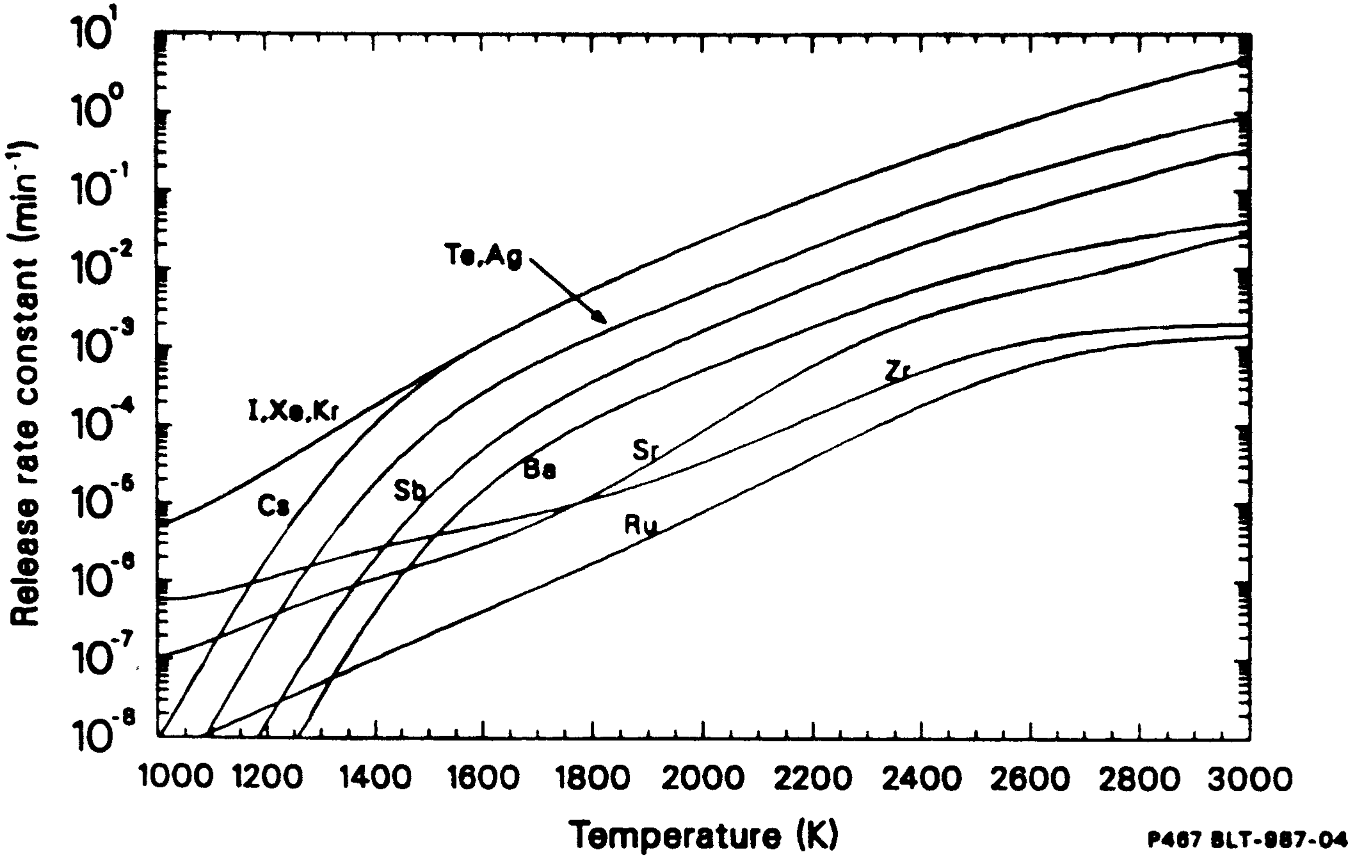

Figure 2. Fission product release correlations from NUREG-0772 (Ref. 6).

TABLE 1. ESTIMATED FISSION PRODUCT RELEASE FOR TMI-2 CORE MATERIAL DURING THE DEGRADED CORE HEATUP PERIOD (100-174 minutes)

Fission Product	Relative Volatility	Approximate Release Rate[a] (fraction/min)	Estimated TMI-2 Release (%)
I, Cs, Xe, Kr	High	0.1	100
Te, Ag	High	0.015	30-40
Sb	Medium	0.007	15-18
Ba	Medium	0.0012	2-3
Sr	Low	0.0008	1.5-2
Ru	Low	0.00002	<1

a. Release rates from Ref. 6 (Fig. 2); estimated time at temperature (2200 K) is assumed to be 20-25 minutes.

Based on these core temperatures and fission product release correlations, most of the highly volatile fission products should have been released from the consolidated regions. However, the lower and peripheral regions of core remained coolable and intact. Calculated total core release of the volatile fission products during this period of the accident, using fuel damage models in which the core temperature gradient was modeled, range from 30 to 40% of the core inventory (Ref. 7).

The release and transport of fission products during this period are complicated by many factors. The more important of these include:

- Uncertainty in coolant flow due to the changing core configuration, particularly the blockage of coolant flow through the center (consolidated) region,

- Uncertainty of when core slumping occurred and the behavior of fission products in the large region of consolidated core material during and after the slumping process,

- Interaction of the relocating molten core materials with the coolant, thus affecting the cooldown of the molten core material, core steam flow, core cooling, and core oxidation rates,

- The effect of control rod material behavior on the fission product transport, chemistry, and core damage progression.

Transport of fission products from the RCS during the initial core heatup and relocation period was limited to the gaseous fission products initially released at the time of cladding rupture and most likely was limited to a relatively small number of rods. Coolant loss from the RCS was limited to the makeup/letdown pathways during the period of significantly increasing fission product release from the fuel between 150-174 minutes.

B-Loop Pump Transient (174-180 minutes). At approximately 174 minutes, the operators turned on the 2B-loop pump. Although the pump continued running for approximately 19 minutes, significant coolant injection into the vessel lasted only momentarily (<1 minute as measured by the steam flow rate in the hot legs). During this brief time, there was sufficient water injected into the vessel to have covered the core. The cooling trend of the incore thermocouples indicate cooling of the core was limited to the peripheral regions.

The impact of the pump transient was significant for two reasons. First, the upper regions of the core were severely fragmented from the thermal/mechanical shock, thus creating the upper core debris as shown in Fig. 3. Second, the liquid carry-over from the lower vessel to the upper vessel regions due to the reflood steam generation would have had a "washing" effect on the upper plenum structures, releasing fission products which were not chemically bound to the upper plenum surfaces.

It has been hypothesized that enhanced fission product release may have occurred during the fuel shattering and debris formation as a result of UO_2 fracturing along the grain boundaries. However, it is not clear from present evaluation of the upper core debris examination data that significant grain boundary fracturing occurred. Additional evaluation of the upper core debris particle examination data is presently underway to address the enhanced release hypothesis.

<u>Degraded Core Heatup (176-224 minutes)</u>. During this period, the interior of the consolidated core material continued to heat up because internal heat generation was larger than the surface heat loss. Consequently, a region of molten core material formed in the interior and continued to grow with time. The best-estimate scenario suggests that by 224 minutes, nearly all of the material in the consolidated core region (approximately 30-40 percent of the original core material), with the exception of a surrounding crust which served to keep the molten material in place, had reached melting temperatures (2800-3100 K) as shown in Fig. 4.

The fission product behavior within the molten pool is thought to have been primarily dependent on the natural circulation (convection flows) and

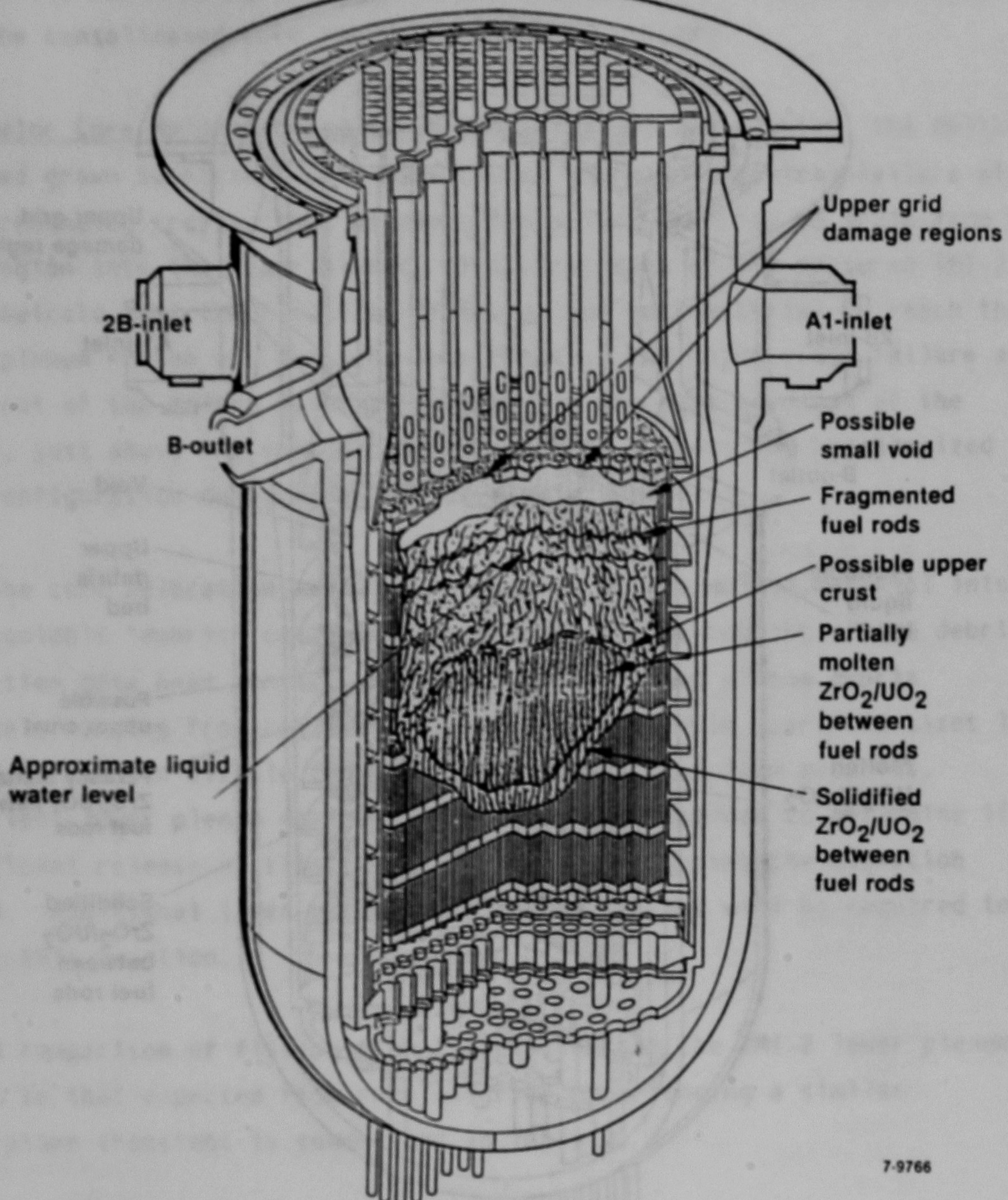

Figure 3. Hypothesized degraded TMI-2 core configuration just after the B-pump transient (175 minutes).

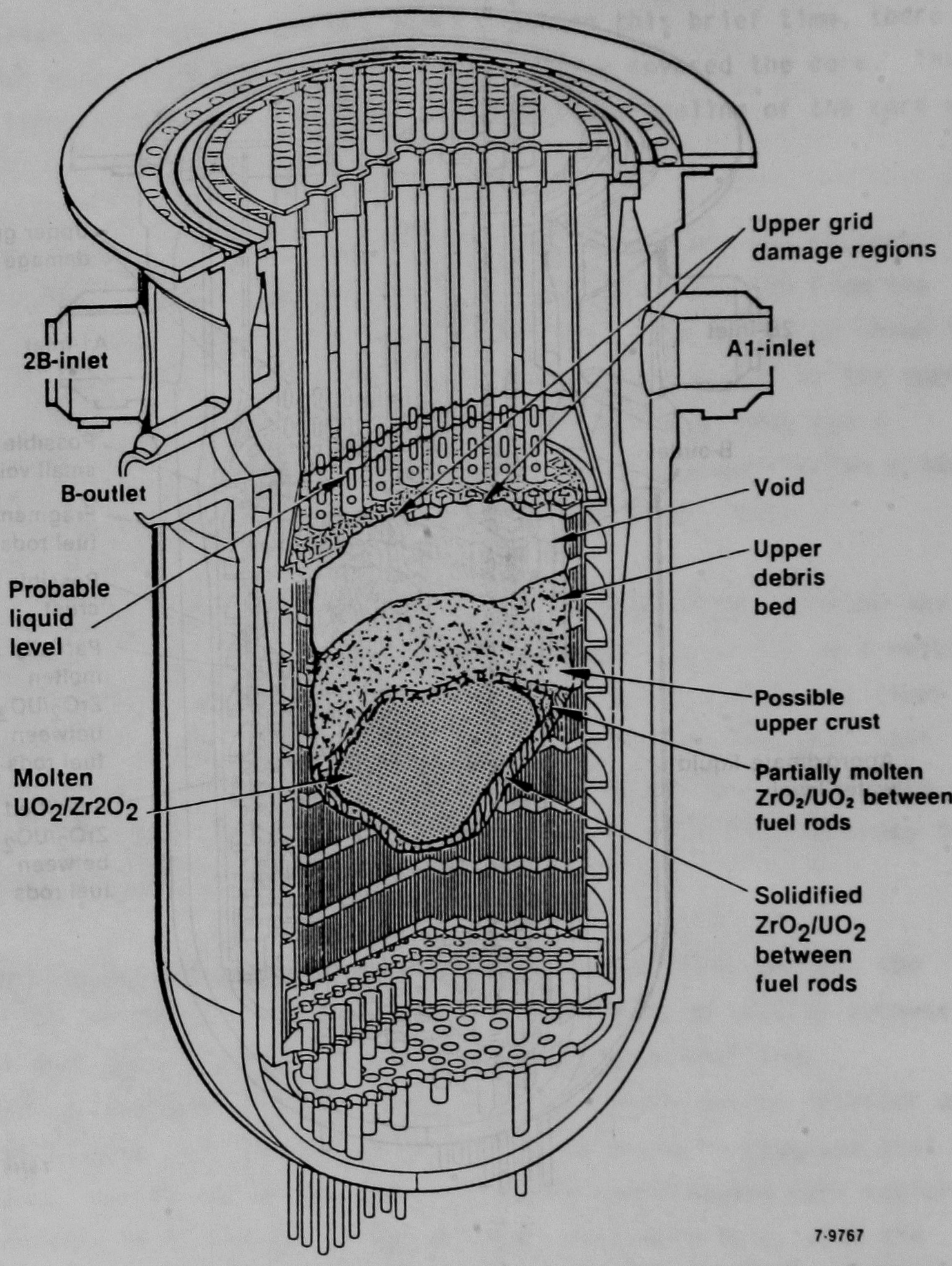

Figure 4. Hypothesized degraded core configuration just before the molten core relocation (224 minutes).

bubble dynamics within the pool and the chemical interactions between the various molten core constituents. Slow diffusion from the molten material through the surrounding crust would have limited fission product release from the consolidated core region during this period.

Major Core Relocation (224-226 minutes). By 224 minutes, the molten pool had grown sufficiently to have caused thermal/mechanical failure of the surrounding crusts, thus allowing the molten material to drain from the core region into the lower plenum. Interpretation of the measured TMI-2 data indicate that the flow time of the molten core material to reach the lower plenum region was less than one minute. The major crust failure and break-out of the molten material occurred in the east quadrant of the vessel, just above the core mid plane. Figure 5 shows the hypothesized core configuration during the 224-226 minute period.

The core relocation resulted in breakup of the molten material into a more coolable "debris" configuration and subsequent cooling of the debris. Inspection data have confirmed a wide range of lower plenum debris material, ranging from relatively fine, uniform debris (particle sizes less than 0.5 inch) to sizable flows of consolidated lava-like material. Sufficient lower plenum debris has not yet been examined to determine if significant release of fission products occurred during the migration period. Additional lower plenum debris examinations will be required to answer this question.

A comparison of fission product release from the TMI-2 lower plenum debris to that expected from core material experiencing a similar temperature transient is summarized in Table 2.

RCS Recovery Period (226 minutes-15.5 hours). This period was characterized by continued attempts to establish forced coolant flow through the vessel. Cooling of the degraded core continued throughout this period. It is estimated the lower plenum debris was cooled within 0.5-1.0 h after the initial relocation at 224 minutes. However, the material in the degraded core region required days to reach equilibrium with the surrounding coolant.

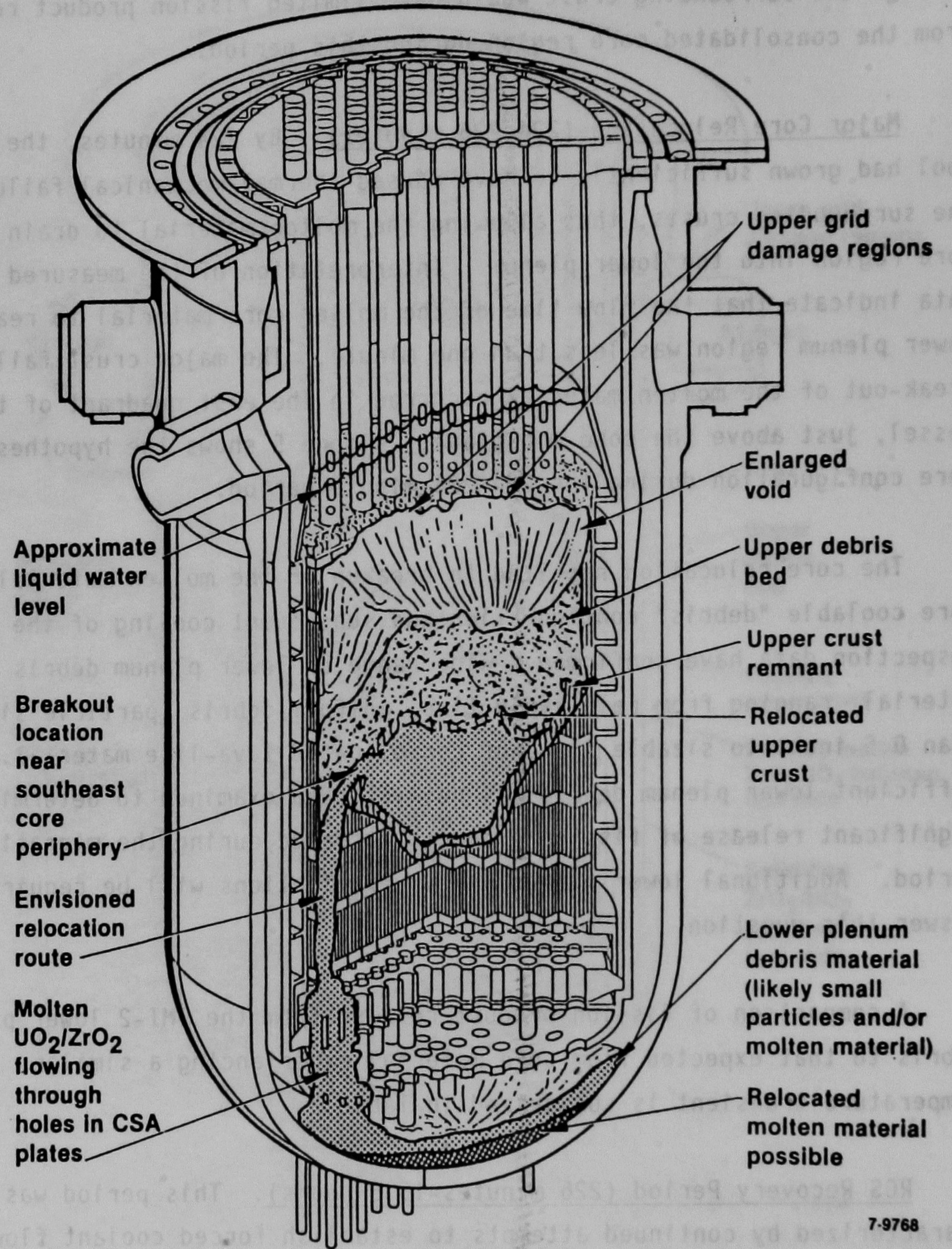

Figure 5. Hypothesized core configuration during core failure and migration period (224-226 minutes).

TABLE 2. COMPARISON OF (a) FISSION PRODUCT RELEASE ESTIMATES FROM THE TMI-2 LOWER PLENUM DEBRIS, AND (b) RELEASE FROM UO_2 FUEL ASSUMED TO BE AT TEMPERATURES RANGING FROM 2800-3100 K FOR 10-30 MINUTES USING THE NUREG-0772 RELEASE CORRELATIONS

Fission Product	TMI-2 Release (%)[a]	NUREG-0772 Release (%)
Cs	65-100	100
Sb	85-100	100
Sr	32-100	10-60
Ru	84-100	1-3

a. The release estimates are based on measured retention values from Ref. 8.

In attempting to establish RCS flows during this period, the pressurizer block valve was cycled many times, resulting in significant coolant transfer to the containment building. Since the majority of fission products were resident in the RCS coolant shortly after the relocation event (226 minutes), the major fission product transport from the RCS to the containment occurred during this time period. The coolant flow through the PORV is shown in Fig. 6; notice that very little transport of RCS water to the containment occurred between the time of pressurizer block value closure (~140 minutes) and the major core relocation at 225 minutes. Thus most of the water transferred from the RCS to the containment (and hence the fission product transport from the RCS) occurred after the major core relocation at 224 minutes.

Measurable Fission Product Isotopes. Table 3 lists the important fission product (and actinide) isotopes commonly considered for evaluating severe accidents. Using the ORIGEN2-calculated total core activity levels (Ref. 10) at the time of shutdown for those isotopes listed in Table 3 and the listed decay constants for each isotopes, those radioisotopes contributing 95% of the total core activity were estimated for times of 1 day, 1 month, 1 year, and approximately 8 years (to present). The results (Table 4) show that the total activity has decayed several orders of magnitude since the accident and those isotopes contributing 95% of the total activity have decreased from 17 at one day to only five at the present time. Thus, only a limited number of radioisotopes can presently be readily measured from the core material samples.

In the next Section we discuss the methodology used to estimate the activity inventory and the resulting inventory estimates for the longer-lived (measurable) isotopes.

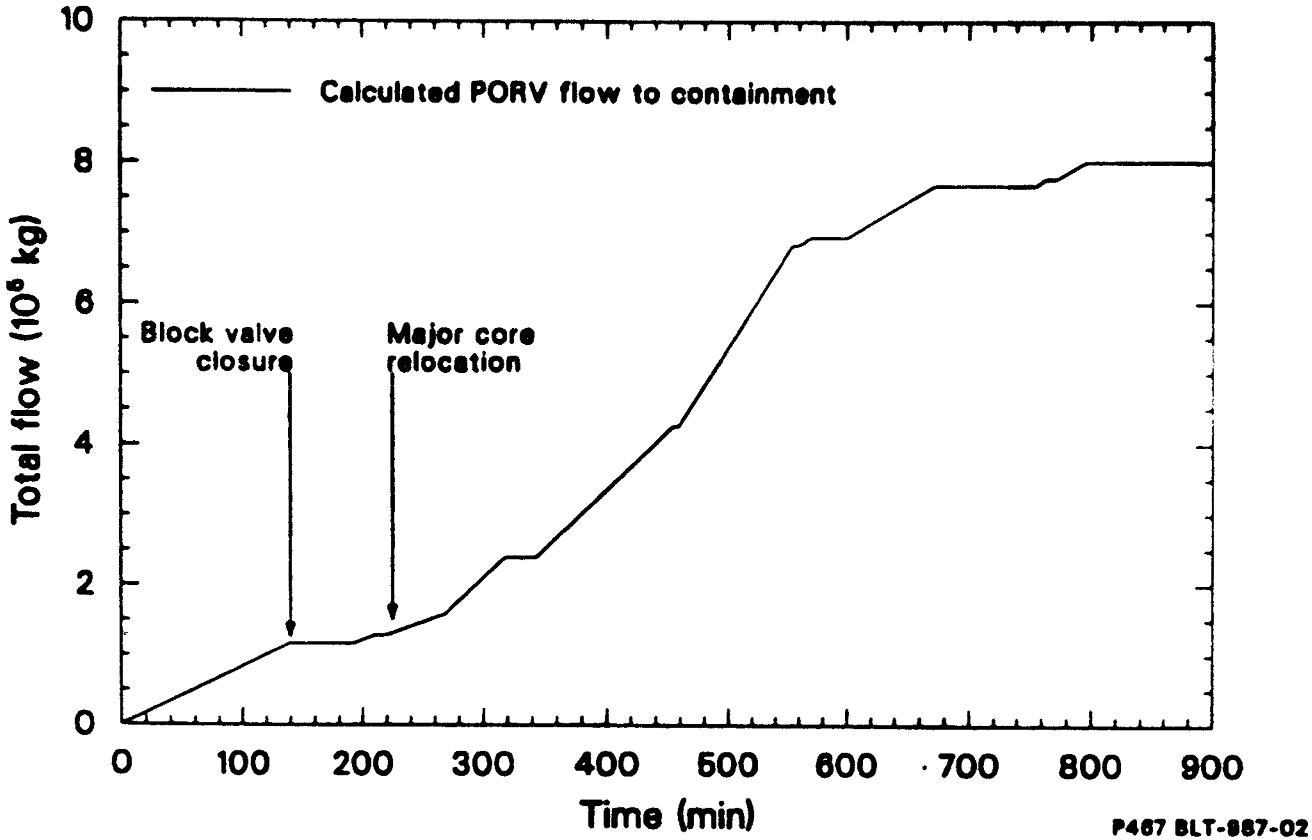

Figure 6. Estimated TMI-2 PORV flow vs. time.

TABLE 3. RADIOISOTOPES CONSIDERED IMPORTANT FOR LIGHT WATER REACTOR SEVERE ACCIDENT SOURCE TERMS

Element	Isotope	Half Life (days)
Cobalt	Co-60	1.92×10^{3}
Cobalt	Co-58	7.10×10^{1}
Krypton	Kr-85	3.95×10^{3}
Krypton	Kr-85M	1.83×10^{-1}
Krypton	Kr-87	5.28×10^{-2}
Krypton	Kr-86	1.17×10^{-1}
Rubidium	Rb-86	1.87×10^{1}
Strontium	Sr-89	5.21×10^{1}
Strontium	Sr-90	1.10×10^{4}
Strontium	Sr-91	4.03×10^{-1}
Yttrium	Yt-90	2.67×10^{0}
Ytrium	Yt-91	5.90×10^{1}
Zirconium	Zr-95	6.52×10^{1}
Zirconium	Zr-97	7.10×10^{-1}
Niobium	Nb-95	3.50×10^{1}
Molybdenum	Mo-99	2.80×10^{0}
Technetium	Tc-99M	2.50×10^{-1}
Ruthenium	Ru-103	3.95×10^{1}
Ruthenium	Ru-105	1.85×10^{-1}
Ruthenium	Ru-106	3.66×10^{2}
Rhodium	Rh-105	1.50×10^{0}
Tellurium	Te-127	3.91×10^{-1}
Tellurium	Te-127	1.09×10^{2}
Tellurium	Te-129	4.80×10^{-2}
Tellurium	Te-129M	3.40×10^{-1}
Tellurium	Te-131M	1.25×10^{0}
Tellurium	Te-132	3.25×10^{0}
Antimony	Sb-126	1.24×10^{1}
Antimony	Sb-125	9.96×10^{2}
Iodine	I-131	8.05×10^{0}
Iodine	I-132	9.58×10^{-2}
Iodine	I-133	8.75×10^{-1}
Iodine	I-134	3.66×10^{-2}
Iodine	I-129	5.80×10^{9}
Xenon	Xe-133	5.28×10^{0}
Xenon	Xe-135	3.84×10^{-1}

TABLE 3. (continued)

Element	Isotope	Half Life (days)
Cesium	Cs-134	7.50×10^{2}
Cesium	Cs-135	8.40×10^{8}
Cesium	Cs-137	1.10×10^{4}
Barium	Ba-140	1.28×10^{1}
Lanthanum	La-140	1.67×10^{0}
Cerium	Ce-141	3.23×10^{1}
Cerium	Ce-143	1.38×10^{0}
Cerium	Ce-144	2.84×10^{2}
Europium	Eu-154	2.99×10^{3}
Euripium	Eu-155	1.74×10^{3}
Neptunium	Np-239	2.35×10^{0}
Plutonium	Pu-238	3.25×10^{4}
Plutonium	Pu-239	8.90×10^{6}
Plutonium	Pu-240	2.40×10^{6}
Plutonium	Pu-241	5.35×10^{3}
Americium	Am-241	1.50×10^{5}
Curium	Cm-242	1.63×10^{2}
Curium	Cm-244	6.63×10^{3}

TABLE 4. TOTAL TMI-2 CORE ACTIVITY LEVEL AND RADIOISOTOPES CONTRIBUTING 95% OF THE TMI-2 TOTAL ACTIVITY VS. TIME AFTER THE ACCIDENT

Time After Accident	Total Core Activity (curies)	Isotopes and Their Fractional Contribution to the Total Core Activity	
1 day	3.1×10^9	Np-239	(0.392)
		Ba-140	(0.049)
		Xe-133	(0.048)
		Zr-95	(0.048)
		Nb-95	(0.047)
		Ce-141	(0.047)
		Mo-99	(0.040)
		Yt-91	(0.038)
		Ru-103	(0.035)
		La-140	(0.034)
		Te-132	(0.031)
		Sr-89	(0.030)
		Ce-144	(0.027)
		Ce-143	(0.025)
		I-131	(0.025)
		I-133	(0.025)
		Zr-97	(0.018)
1 month	6.4×10^8	Zr-95	(0.168)
		Yt-91	(0.130)
		Nb-95	(0.128)
		Ce-144	(0.122)
		Ce-141	(0.121)
		Ru-103	(0.100)
		Sr-89	(0.097)
		Ba-140	(0.047)
		Ru-106	(0.036)
1 year	7.1×10^7	Ce-144	(0.490)
		Ru-106	(0.176)
		Cs-134	(0.075)
		Cs-137	(0.064)
		Sr-90	(0.051)
		Pu-241	(0.046)
		Zr-95	(0.044)
		Yt-91	(0.023)

TABLE 4. (continued)

Time After Accident	Total Core Activity (curies)	Isotopes and Their Fractional Contribution to the Total Core Activity	
100 months (To 8/87)	9.6×10^6	Cs-137	(0.389)
		Sr-90	(0.306)
		Pu-241	(0.222)
		Kr-85	(0.031)
		Cs-134	(0.028)

3. FISSION PRODUCT INVENTORY

Three steps are necessary to estimate the post-accident TMI-2 fission product inventory for any given radioisotope. The first step is to identify the major TMI-2 fission product repositories. These repositories can be grouped into two general categories, internal to the reactor coolant system (RCS) and external to the RCS. Recovery work over the past several years has identified the major ex-vessel repositories. Recent work has provided the data necessary to define the end-state condition of the core and locations of the degraded core material. These unique regions of degraded core material complete our knowledge of the major fission product repositories. Table 5 lists the major TMI-2 fission product repositories.

Of special interest are the degraded core regions, since they have only recently been characterized and contain the majority of fission products (except for the highly volatile species). The degraded core material is distributed within the reactor vessel in six major regions as noted in the Table 6 and shown in Fig. 7.

The second step in estimating a specific isotopic inventory is to characterize the material in each repository. This characterization includes (a) estimating the mass of radioactive material contained within each repository, (b) acquisition of representative samples from each major repository, and (c) examination of the material to quantify specific isotopic activity levels (μCi/g). With the exception of the degraded core regions, this characterization work has nearly been completed. Samples have been, or will be, acquired from each unique degraded core region as the core is defueled and examination data are expected within the next 18 months. For the degraded core zones containing previously molten core material (molten core, core former zone, and CSA regions), the specific activities (μCi/g) as measured from the previously molten lower plenum debris particles <u>are assumed</u> to be applicable for the inventory estimates.

Measurements of the radioactivity from the partially intact fuel rods are not yet available. However, estimates of the core fission products

TABLE 5. MAJOR TMI-2 FISSION PRODUCT REPOSITORIES

In-RCS Repositories		Ex-RCS Repositories
Degraded Core Regions	Reactor Cooling System	
Upper core debris	Hot leg piping surfaces	Reactor building water
Previously molten core material	Upper plenum surfaces	Reactor building sediment
	Steam generator surfaces	
Partially intact fuel rods	Pressurizer surfaces	Reactor building lower walls
Previously molten core material in the core former zone[a]	Steam generator sediment	Reactor building upper surfaces
	Pressurizer sediment	
Previously molten core material in the core support assembly (CSA) region	Makeup/purification demineralizer sediment	Reactor building air space
	Reactor coolant drain tank	Auxiliary building liquid
Lower plenum debris	RCS coolant	Auxiliary building gas release

a. Core former zone includes that region containing the core former plates and baffle plates.

TABLE 6. END-STATE CORE REGION MASSES[a]

Region	Estimated Mass kg	(uncertainty)
Upper core debris	23,700	(±1,200)
Previously molten material in the core region	25,600	(±6,600)
Intact rod stubs	47,000	(±3,800)
Previously molten core material in the core former region	5,000	(±2,000)
Previously molten core material in the core support assembly	5,000	(±2,000)
Lower plenum debris	15,000	(±5,000)

a. The mass of the original core material (including structural components) is approximately 125,000 kg (Ref. 1).

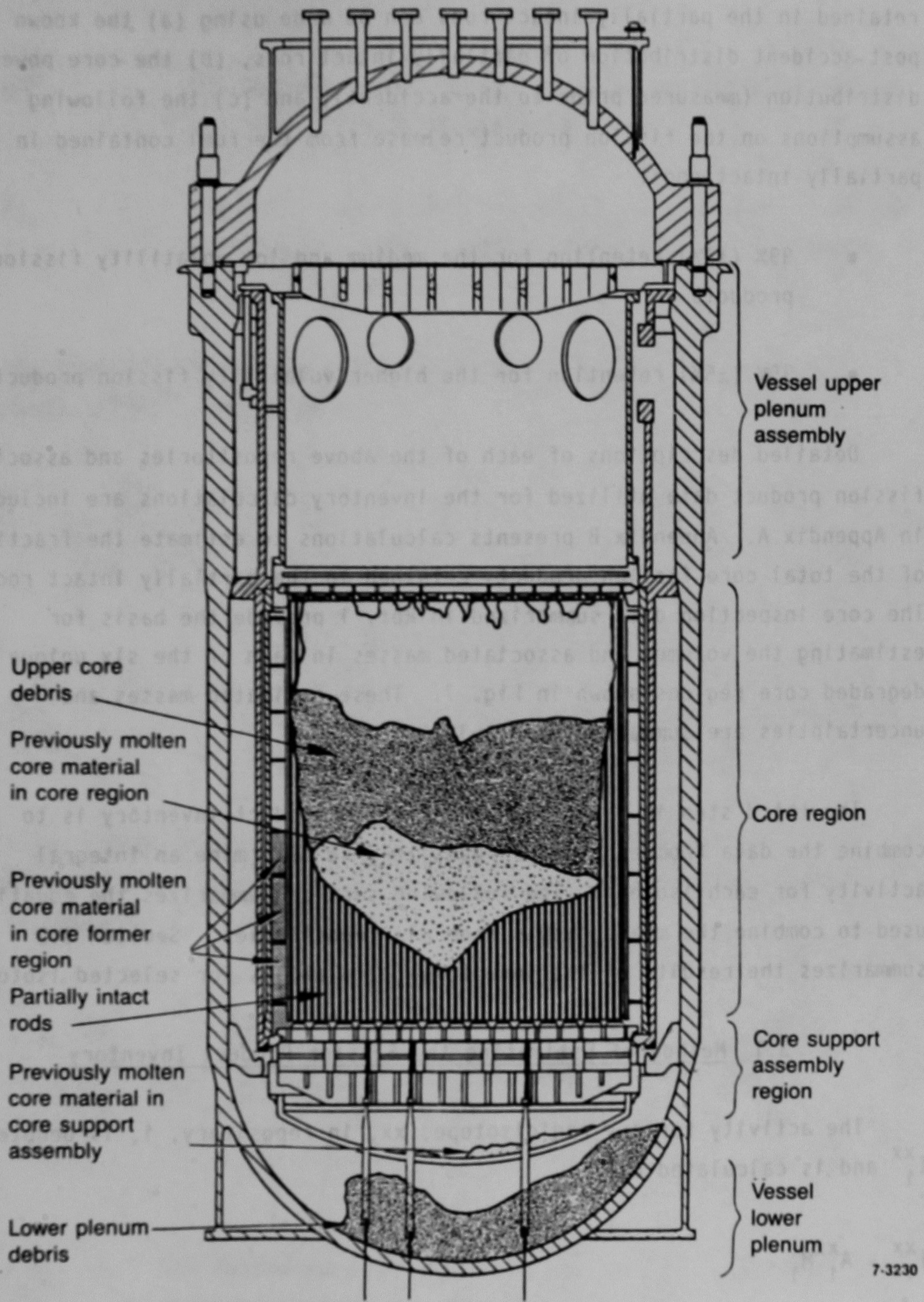

Figure 7. End-state degraded core regions (core cross-section through row K of fuel assemblies).

retained in the partially intact rods can be made using (a) the known post-accident distribution of partially intact rods, (b) the core power distribution (measured prior to the accident), and (c) the following assumptions on the fission product release from the fuel contained in the partially intact rods:

- 99% (±1%) retention for the medium and low volatility fission products
- 95% (±5%) retention for the higher volatility fission products.

Detailed descriptions of each of the above repositories and associated fission product data utilized for the inventory calculations are included in Appendix A. Appendix B presents calculations to estimate the fraction of the total core fission products retained in the partially intact rods. The core inspection data summarized in Ref. 1 provide the basis for estimating the volumes and associated masses in each of the six unique degraded core regions shown in Fig. 7. These estimated masses and uncertainties are summarized in Table 6.

The third step in estimating the fission product inventory is to combine the data from each major repository to determine an integral activity for each isotope. The following section summarizes the equations used to combine the measurements from all repositories. Section 3.2 summarizes the results of the inventory calculations for selected isotopes.

3.1 Method of Estimating the Fission Product Inventory

The activity for any radioisotope, xx, in repository, i, is denoted by I_i^{xx} and is calculated by,

$$I_i^{xx} = A_i^{xx} M_i \qquad (1)$$

where

A_i^{xx} = The average isotopic activity from the material within the repository (μCi/sample unit), and

M_i = the total sample units of radioactive material from the repository. Sample units include: mass (g); volume (ml); and surface area (cm_2).

The fraction of the total core inventory accounted for in each repository, $FRAC_i^{xx}$, can be defined as

$$FRAC_i^{xx} = \frac{I_i^{xx}}{I_{\text{Total Core}}^{xx}} \tag{2}$$

where

$I_{\text{Total Core}}^{xx}$ = is the total core inventory as estimated from the ORIGEN2 calculations (Ref. 9).

Using linear error propagation techniques (Refs. 10 and 11), the uncertainty[a] in the total isotopic activity, for any given repository, σ_i^{xx}, is given by,

$$\sigma_i^{xx} = \left[\left(\frac{\partial I_i^{xx}}{\partial A_i^{xx}}\right)^2 \sigma_{A_i^{xx}}^2 + \left(\frac{\partial I_i^{xx}}{\partial M_i}\right)^2 \sigma_{M_i}^2\right]^{1/2} = \left[(M_i)^2\, \sigma_{A_i^{xx}}^2 + \left(A_i^{xx}\right)^2 \sigma_{M_i}^2\right]^{1/2} \tag{3}$$

where

$\sigma_{A_i^{xx}}$ = the estimated uncertainty in the isotopic activity measurement from the ith repository, and

a. All uncertainty values are expressed as 95% confidence estimates.

σ_{M_i} = the estimated uncertainty in the $\underline{i}$th repository mass (sample unit).

Estimating the total TMI-2 isotopic activity, I_T^{xx}, requires summing individual repository contributions, i.e.,

$$I_T^{xx} = \sum_{i=1}^{n} I_i^{xx} \quad . \tag{4}$$

In addition, the uncertainty in the total activity estimate, $\sigma_{I_T^{xx}}$, is given by,

$$\sigma_{I_T^{xx}} = \left[\sum_{i=1}^{n} \left(\sigma_i^{xx}\right)^2 \right]^{1/2} \quad . \tag{5}$$

Error propagation theory also allows estimation of the fractional contribution to the total uncertainty from each repository, F_i^{xx}. This parameter is useful for identifying the major contributors to the total uncertainty, thus indicating where additional data would reduce the inventory uncertainty. The fractional uncertainty contribution from each repository is calculated using the following expression,

$$F_i^{xx} = \frac{\left(\sigma_i^{xx}\right)^2}{\left(\sigma_{I_T^{xx}}\right)^2} \quad . \tag{6}$$

Finally, the fraction of the total inventory accounted for from all repositories, $FRAC_{Total}^{xx}$, is estimated by taking the ratio of the total activity from all repositories compared to the original core inventory as calculated by ORIGEN2 (Ref. 9), i.e.,

$$FRAC_{Total}^{xx} = \frac{I_T^{xx}}{I_{Total\ Core}^{xx}} \tag{7}$$

and the associated uncertainty in the fraction of the total inventory accounted for, $\sigma_{FRAC^{xx}_{Total}}$, is

$$\sigma_{FRAC^{xx}_{Total}} = \frac{I^{xx}_{T}}{I^{xx}_{Total\ Core}} \quad . \tag{8}$$

3.2 Summary of Inventory Calculations

The above equations were used to determine best-estimate fractional core inventories and associated uncertainties for the following radioisotopes:

Low Volatility:	Krypton (Kr-85)
	Cesium (Cs-137)
	Iodine (I-129)
Medium Volatility:	Antimony (Sb-125)
	Strontium (Sr-90)
Low Volatility:	Ruthenium (Ru-106)
	Cerium (Ce-144)
	Europium (Eu-144)

For each isotope, a calculational data sheet is included that summarizes the following:

1. Initial total core inventory, I^{xx}_{Total} (μCi).

2. Decay half-life.[a]

3. Total decay factor since the accident.

---

a. All calculated inventory values are decay corrected to July 1, 1987.

4. Measured sample specific activity (A_i^{xx} in Eq. 1). Uncertainty in the measured values are shown in parenthesis adjacent to the nominal values.[a]

5. Mass (volume/area) of material associated with each repository sample data (M_i in Eq. 1).

6. Repository inventory, I_i^{xx}, (μCi) and calculated uncertainty, σ_i^{xx}, (Eqs. 1 and 3).

7. Fractional core inventory accounted for from each repository, $FRAC_i^{xx}$, (from Eq. 2).

8. Repository fractional contribution to the uncertainty in total core inventory, F_i^{xx}, (from Eq. 6).

9. Reference Table number in Appendix A summarizing the data.

10. Total core inventory from all repositories, I_T^{xx}, and associated uncertainty (from Eqs. 4 and 5)

11. Total fractional core inventory accounted for and associated uncertainty (from Eqs. 7 and 8).

3.2.1 Estimated Noble Gas Inventory (Kr-85)

Table 7 summarizes the Kr-85 inventory estimates. As seen from the table, the activity in the containment air space accounts for approximately 54% of the original core inventory. Another 31% is estimated to be in the intact fuel rods.

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a. In some cases, uncertainty values for the activity measurements or estimated masses of degraded core material are not documented. For these cases, the uncertainty is assumed to be zero. Realistic estimates for these uncertainties will be evaluated during the fission product data qualification process and used in the final inventory calculations.

TABLE 7. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Kr-85[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (µCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	NM[c]					
Molten Core Zone	NM					
Intact Rods	9.2×10^{10} (4.9×10^{9}) µCi	3.3×10^{-1} (2.5×10^{-2})	1.8×10^{10} (1.7×10^{9})	0.310	0.582	A-3
Core Former Region	NM					
Core Support Assembly	NM					
Lower Plenum Debris	NM					
Reactor Cooling System						
Hot Leg surfaces	NM					
Upper Plenum Surfaces	NM					
SG Surfaces	NM					
Pressurizer Surfaces	NM					
SG Sediment	NM					
Pressurizer Sediment	NM					
Domin Sediment	NM					
RCDT Sediment	NM					
RCS Coolant	NM					
Outside RCS						
R/B[d] Liquid	NM					
R/B Sediment	NM					
R/B Lower Walls	NM					
R/B Upper Surfaces	NM					
R/B Air Space	8.8×10^{-1} (4.0×10^{-2}) µCi/cm^3	5.6×10^{10} ( ) cm^3	3.1×10^{10} (1.4×10^{9})	0.540	0.418	A-21
Aux/B[e] Liquid	NM					
Aux/B Gas Release	9.7×10^{8} ( ) µCi	1.0×10^{0} ( )	5.8×10^{8} ( )	0.010	0.000	A-23

a. All decay data corrected to July 1987.
Half Life = 3950 days
Decay Factor Since Accident = 0.5939

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured.

d. R/B = Reactor Building.

e. Aux/B = Auxiliary Building.

Total Core Inventory[a]	=	5.8×10^{10} µCi
Inventory Accounted for to July 1987	=	5.0×10^{10} ($\pm 2.2 \times 10^{9}$) µCi
Fraction Accounted for	=	0.860 (± 0.038)

Measurements have not been made to determine the retained Kr-85 activity levels in the degraded core materials. Examinations are presently underway to determine these values for the upper debris bed (2 samples), previously molten core material (2 core bore samples), the lower plenum debris (2 samples), and the intact fuel rods (2 samples from upper level rods, 6 samples from lower rod stubs).

3.2.2 Estimated Inventory of High Volatility Fission Products (Cesium and Iodine)

Table 8 summarizes the calculated inventory estimates for Cs-137. As seen from the table, the Cs-137 can be accounted for in the degraded core materials and the reactor basement (concrete walls and liquid). Minor amounts (<2%) were measured in the RCS water and the makeup and letdown demineralizer sediment.

Table 9 summarizes the calculated inventory estimates for I-129. The total iodine inventory estimate is approximately 68%. Several sources of uncertainty exist and will require additional evaluation. These include reactor building basement sediment, reactor building basement liquid, and RCS coolant. Data from these repositories will be evaluated further during the fission product data qualification process.

3.2.3 Estimated Inventory of the Medium Volatility Fission Products (Antimony and Strontium)

Table 10 summarizes the inventory estimates for Sb-125. An estimated 43% of the antimony has been accounted for and most of this is predicted to be in the intact rods. Thus, additional core material examinations (including intact rod segments) will be important in improving the estimated antimony inventory.

Table 11 summarizes the estimated inventory for Sr-90. The estimated strontium inventory appears to account for the ORIGEN2 calculated total core activity.

TABLE 8. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Cs-137[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (µCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	1.6×10^{3} (1.0×10^{1}) µCi/g	2.4×10^{7} (1.2×10^{6}) g	3.4×10^{10} (1.7×10^{9})	0.048	0.006	A-1
Molten Core Zone	8.7×10^{2} (2.0×10^{0}) µCi/g	2.6×10^{7} (6.6×10^{6}) g	2.2×10^{10} (5.6×10^{9})	0.031	0.064	A-2
Intact Rods	8.1×10^{11} (4.3×10^{10}) µCi	3.3×10^{-1} (2.5×10^{-2})	2.2×10^{11} (2.0×10^{10})	0.309	0.857	A-3
Core Former Region	8.7×10^{2} (2.0×10^{0}) µCi/g	5.0×10^{6} (2.0×10^{6}) g	4.2×10^{9} (1.7×10^{9})	0.006	0.006	A-4
Core Support Assembly	8.7×10^{2} (2.0×10^{0}) µCi/g	5.0×10^{6} (2.0×10^{6}) g	4.2×10^{9} (1.7×10^{9})	0.006	0.006	A-5
Lower Plenum Debris	8.7×10^{2} (2.0×10^{0}) µCi/g	1.5×10^{7} (5.0×10^{6}) g	1.3×10^{10} (4.2×10^{9})	0.018	0.037	A-6
Reactor Cooling System						
Hot Leg surfaces	2.0×10^{1} (1.0×10^{-1}) µCi/cm²	9.7×10^{5} (9.7×10^{4}) cm²	1.8×10^{7} (1.8×10^{6})	0.000	0.000	A-7
Upper Plenum Surfaces	6.6×10^{1} (6.5×10^{0}) µCi/cm²	3.5×10^{6} (7.1×10^{5}) cm²	2.1×10^{8} (4.8×10^{7})	0.000	0.000	A-8
SG Surfaces	3.6×10^{0} (1.0×10^{-2}) µCi/cm²	3.7×10^{7} (3.7×10^{6}) cm²	1.2×10^{8} (1.2×10^{7})	0.000	0.000	A-9
Pressurizer Surfaces	4.2×10^{-2} (3.0×10^{-4}) µCi/cm²	1.0×10^{6} (1.0×10^{5}) cm²	3.9×10^{4} (3.9×10^{3})	0.000	0.000	A-10
SG Sediment	2.6×10^{3} ( ) µCi/g	8.0×10^{3} (5.0×10^{3}) g	2.1×10^{7} (1.3×10^{7})	0.000	0.000	A-11
Pressurizer Sediment	3.7×10^{1} ( ) µCi/g	6.6×10^{4} ( ) g	2.4×10^{6} ( )	0.000	0.000	A-12
Demin Sediment	1.4×10^{4} ( ) µCi/g	9.3×10^{5} ( ) g	1.2×10^{10} ( )	0.016	0.000	A-13
RCBT Sediment	9.7×10^{1} (2.0×10^{0}) µCi/g	2.6×10^{4} (5.2×10^{3}) g	2.3×10^{6} (4.7×10^{5})	0.000	0.000	A-14
RCS Coolant	3.1×10^{1} (2.0×10^{-1}) µCi/ml	3.3×10^{8} ( ) ml	8.7×10^{9} (5.7×10^{7})	0.012	0.000	A-15
Outside RCS						
R/B[c] Liquid	1.4×10^{2} (1.0×10^{0}) µCi/ml	2.4×10^{9} ( ) ml	3.0×10^{11} (2.1×10^{9})	0.420	0.009	A-17
R/B Sediment	8.1×10^{2} (3.0×10^{0}) µCi/g	3.8×10^{5} ( ) g	2.6×10^{8} (9.8×10^{5})	0.000	0.000	A-18
R/B Lower Walls	3.0×10^{3} ( ) µCi/cm²	2.9×10^{7} ( ) cm²	8.3×10^{10} (0.00×10^{0})	0.118	0.000	A-19
R/B Upper Surfaces	7.6×10^{-1} (8.0×10^{-1}) µCi/cm²	1.9×10^{8} ( ) cm²	1.3×10^{8} (1.4×10^{8})	0.000	0.000	A-20
R/B Air Space	7.2×10^{-10} (8.0×10^{-11}) µCi/cm³	5.6×10^{10} ( ) cm³	3.4×10^{1} (3.8×10^{0})	0.000	0.000	A-21
Aux/B[d] Liquid	4.4×10^{1} (4.0×10^{-1}) µCi/ml	9.5×10^{8} (7.0×10^{7}) ml	3.6×10^{10} (2.7×10^{9})	0.051	0.015	A-22
Aux/B Gas Release	NM[e]					

a All decay data corrected to July 1987.
Half Life = 11000 days
Decay Factor Since Accident = 0.8294

b Where no uncertainty is shown, it is unknown.

c. R/B = Reactor Building.

d. Aux/B = Auxiliary Building.

e. NM = Not Measured.

Total Core Inventory[a]	=	7.0×10^{11} µCi
Inventory Accounted for to July 1987	=	7.3×10^{11} ($\pm 2.2 \times 10^{10}$) µCi
Fraction Accounted for	=	1.036 (±0.031)

TABLE 9. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF I-129[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (μCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	4.4×10^{-4} ( ) μCi/g	2.4×10^{7} (1.2×10^{6}) g	1.0×10^{4} (5.2×10^{2})	0.048	0.005	A-1
Molten Core Zone	5.6×10^{-5} (3.2×10^{-6}) μCi/g	2.6×10^{7} (6.6×10^{6}) g	1.4×10^{3} (3.8×10^{2})	0.007	0.003	A-2
Intact Rods	2.1×10^{5} (1.1×10^{4}) μCi	3.3×10^{-1} (2.5×10^{-2})	6.7×10^{4} (6.2×10^{3})	0.309	0.695	A-3
Core Former Region	5.6×10^{-5} (3.2×10^{-6}) μCi/g	5.0×10^{6} (2.0×10^{6}) g	2.8×10^{2} (1.1×10^{2})	0.001	0.000	A-4
Core Support Assembly	5.6×10^{-5} (3.2×10^{-6}) μCi/g	5.0×10^{6} (2.0×10^{6}) g	2.8×10^{2} (1.1×10^{2})	0.001	0.000	A-5
Lower Plenum Debris	5.6×10^{-5} (3.2×10^{-6}) μCi/g	1.5×10^{7} (5.0×10^{6}) g	8.4×10^{2} (2.8×10^{2})	0.004	0.001	A-6
Reactor Cooling System						
Hot Leg surfaces	NM[c]					
Upper Plenum Surfaces	5.4×10^{-5} (3.1×10^{-6}) μCi/cm²	3.5×10^{6} (7.1×10^{5}) cm²	1.9×10^{2} (4.0×10^{1})	0.001	0.000	A-8
SG Surfaces	2.9×10^{-6} (2.0×10^{-8}) μCi/cm²	3.7×10^{7} (3.7×10^{6}) cm²	1.1×10^{2} (1.1×10^{1})	0.000	0.000	A-9
Pressurizer Surfaces	NM					
SG Sediment	NM					
Pressurizer Sediment	NM					
Demin Sediment	NM					
RCDT Sediment	5.2×10^{-8} (4.0×10^{-9}) μCi/g	2.6×10^{4} (4.2×10^{3}) g	1.4×10^{-3} (2.9×10^{-4})	0.000	0.000	A-14
RCS Coolant	7.1×10^{-6} (3.0×10^{-7}) μCi/ml	3.3×10^{8} ( ) ml	2.4×10^{3} (10.0×10^{1})	0.011	0.000	A-15
Outside RCS						
R/B[d] Liquid	4.3×10^{-6} (3.0×10^{-7}) μCi/ml	2.4×10^{9} ( ) ml	1.0×10^{4} (7.2×10^{2})	0.048	0.009	A-17
R/B Sediment	1.1×10^{-1} (1.0×10^{-2}) μCi/g	3.8×10^{5} ( ) g	4.1×10^{4} (3.8×10^{3})	0.191	0.254	A-18
R/B Lower Walls	NM					
R/B Upper Surfaces	5.6×10^{-6} (2.2×10^{-6}) μCi/cm²	1.9×10^{8} ( ) cm²	1.0×10^{3} (4.2×10^{2})	0.005	0.003	A-20
R/B Air Space	5.7×10^{-11} (4.0×10^{-12}) μCi/cm³	5.6×10^{10} ( ) cm³	3.2×10^{0} (2.2×10^{-1})	0.000	0.000	A-21
Aux/B[e] Liquid	1.2×10^{-5} (1.0×10^{-6}) μCi/ml	9.5×10^{8} (7.0×10^{7}) ml	1.1×10^{4} (1.3×10^{3})	0.053	0.029	A-22
Aux/B Gas Release	4.3×10^{-2} ( ) μCi	1.0×10^{0} ( )	4.3×10^{-2} ( )	0.000	0.000	A-23

Total Core Inventory[a]	=	2.2×10^{5} μCi
Inventory Accounted for to July 1987	=	1.5×10^{5} ($\pm 7.5 \times 10^{3}$) μCi
Fraction Accounted for	=	0.678 (±0.035)

a. All decay data corrected to July 1987.
Half Life = 5.8×10^{9} days
Decay Factor Since Accident = 1.0000

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured.

d. R/B = Reactor Building.

e. Aux/B = Auxiliary Building.

TABLE 10. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Sb-125[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (µCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	1.0×10^{2} (1.0×10^{0}) µCi/g	2.4×10^{7} (1.2×10^{6}) g	1.1×10^{9} (5.6×10^{7})	0.071	0.019	A-1
Molten Core Zone	8.0×10^{0} (1.3×10^{0}) µCi/g	2.6×10^{7} (6.6×10^{6}) g	1.5×10^{8} (4.5×10^{7})	0.010	0.013	A-2
Intact Rods	1.2×10^{11} (1.2×10^{9}) µCi	3.3×10^{-1} (2.5×10^{-2})	5.1×10^{9} (4.0×10^{8})	0.332	0.960	A-3
Core Former Region	8.0×10^{0} (1.3×10^{0}) µCi/g	5.0×10^{6} (2.0×10^{6}) g	2.9×10^{7} (1.3×10^{7})	0.002	0.001	A-4
Core Support Assembly	8.0×10^{0} (1.3×10^{0}) µCi/g	5.0×10^{6} (2.0×10^{6}) g	2.9×10^{7} (1.3×10^{7})	0.002	0.001	A-5
Lower Plenum Debris	8.0×10^{0} (1.3×10^{0}) µCi/g	1.5×10^{7} (5.0×10^{6}) g	8.7×10^{7} (3.2×10^{7})	0.006	0.006	A-6
Reactor Cooling System						
Hot Leg surfaces	1.3×10^{-1} (1.5×10^{-2}) µCi/cm²	9.7×10^{5} (9.7×10^{4}) cm²	5.3×10^{4} (8.1×10^{3})	0.000	0.000	A-7
Upper Plenum Surfaces	8.9×10^{0} (1.9×10^{0}) µCi/cm²	3.5×10^{6} (7.1×10^{5}) cm²	1.3×10^{7} (3.8×10^{6})	0.001	0.000	A-8
SG Surfaces	1.8×10^{-1} (1.8×10^{-3}) µCi/cm²	3.7×10^{7} (3.7×10^{6}) cm²	2.9×10^{6} (2.9×10^{5})	0.000	0.000	A-9
Pressurizer Surfaces	8.9×10^{-4} (1.3×10^{-4}) µCi/cm²	1.0×10^{6} (1.0×10^{5}) cm²	3.9×10^{2} (6.8×10^{1})	0.000	0.000	A-10
SG Sediment	NM[c]					
Pressurizer Sediment	5.1×10^{-1} ( ) µCi/g	6.6×10^{4} ( ) g	2.3×10^{4} ( )	0.000	0.000	A-12
Demin Sediment	NM					
RCDT Sediment	1.6×10^{1} (8.8×10^{-1}) µCi/g	2.6×10^{4} (5.2×10^{3}) g	1.8×10^{5} (3.7×10^{4})	0.000	0.000	A-14
RCS Coolant	5.1×10^{-2} ( ) µCi/ml	3.3×10^{8} (3.3×10^{8}) ml	3.0×10^{6} ( )	0.000	0.000	A-15
Outside RCS						
R/B[d] Liquid	3.0×10^{-2} (3.0×10^{-3}) µCi/ml	2.4×10^{9} ( ) ml	1.5×10^{7} (1.5×10^{6})	0.001	0.000	A-17
R/B Sediment	4.9×10^{2} (9.8×10^{0}) µCi/g	3.8×10^{5} ( ) g	3.9×10^{7} (7.2×10^{5})	0.003	0.000	A-18
R/B Lower Walls	NM					
R/B Upper Surfaces	NM					
R/B Air Space	2.0×10^{-10} ( ) µCi/cm³	5.6×10^{10} ( ) cm³	1.8×10^{0} ( )	0.000	0.000	A-21
Aux/B[e] Liquid	NM					
Aux/B Gas Release	NM					

Total Core Inventory[a]	=	1.5×10^{10} µCi
Inventory Accounted for to July 1987	=	6.6×10^{9} $(\pm 4.1 \times 10^{8})$ µCi
Fraction Accounted for	=	0.426 (±0.026)

a. All decay data corrected to July 1987.
Half Life = 986 days
Decay Factor Since Accident = 0.1266

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured.

d. R/B = Reactor Building

e. Aux/B = Auxiliary Building.

TABLE 11. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Sr-90[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (μCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	5.6×10^3 (1.0×10^2) μCi/g	2.4×10^7 (1.2×10^6) g	1.2×10^{11} (6.6×10^9)	0.199	0.011	A-1
Molten Core Zone	7.1×10^3 (1.7×10^2) μCi/g	2.6×10^7 (6.6×10^6) g	1.8×10^{11} (4.6×10^{10})	0.284	0.528	A-2
Intact Rods	7.5×10^{11} (7.5×10^9) μCi	3.3×10^{-1} (2.5×10^{-2})	2.0×10^{11} (1.6×10^{10})	0.326	0.062	A-3
Core Former Region	7.1×10^3 (1.7×10^2) μCi/g	5.0×10^6 (2.0×10^6) g	3.4×10^{10} (1.4×10^{10})	0.056	0.048	A-4
Core Support Assembly	7.1×10^3 (1.7×10^2) μCi/g	5.0×10^6 (2.0×10^6) g	3.4×10^{10} (1.4×10^{10})	0.056	0.048	A-5
Lower Plenum Debris	7.1×10^3 (1.7×10^2) μCi/g	1.5×10^7 (5.0×10^6) g	1.0×10^{11} (3.4×10^{10})	0.167	0.302	A-6
Reactor Cooling System						
Hot Leg surfaces	9.4×10^0 (2.0×10^{-1}) μCi/cm²	9.7×10^5 (9.7×10^4) cm²	8.5×10^6 (8.6×10^5)	0.000	0.000	A-7
Upper Plenum Surfaces	2.0×10^1 (2.0×10^{-1}) μCi/cm²	3.5×10^6 (7.1×10^5) cm²	6.5×10^7 (1.3×10^7)	0.000	0.000	A-8
SG Surfaces	1.3×10^{-1} (9.0×10^{-3}) μCi/cm²	3.7×10^7 (3.7×10^6) cm²	4.3×10^6 (5.3×10^5)	0.000	0.000	A-9
Pressurizer Surfaces	3.6×10^{-1} (1.9×10^{-2}) μCi/cm²	1.0×10^6 (1.0×10^5) cm²	3.3×10^5 (3.8×10^4)	0.000	0.000	A-10
SG Sediment	NM[c]					
Pressurizer Sediment	NM					
Demin Sediment	2.0×10^3 ( ) μCi/g	9.3×10^5 ( ) g	1.7×10^9 ( )	0.003	0.000	A-13
RCDT Sediment	1.4×10^3 (7.0×10^2) μCi/g	2.6×10^4 (5.2×10^3) g	3.4×10^8 (6.9×10^7)	0.001	0.000	A-14
RCS Coolant	2.4×10^1 (7.0×10^{-1}) μCi/ml	3.3×10^8 ( ) ml	6.7×10^9 (2.0×10^8)	0.011	0.000	A-15
Outside RCS						
R/B[d] Liquid	5.2×10^0 (3.0×10^{-1}) μCi/ml	2.4×10^9 ( ) ml	1.1×10^{10} (6.2×10^8)	0.017	0.000	A-17
R/B Sediment	8.0×10^2 (2.0×10^2) μCi/g	3.8×10^5 ( ) g	2.6×10^8 (6.5×10^7)	0.000	0.000	A-18
R/B Lower Walls	6.8×10^1 ( ) μCi/cm²	2.9×10^7 ( ) cm²	1.9×10^9 ( )	0.003	0.000	A-19
R/B Upper Surfaces	1.8×10^{-2} (1.7×10^{-2}) μCi/cm²	1.9×10^8 ( ) cm²	3.2×10^6 (3.0×10^6)	0.000	0.000	A-20
R/B Air Space	1.6×10^{-10} (3.0×10^{-11}) μCi/cm³	5.6×10^{10} ( ) cm³	7.6×10^0 (1.4×10^0)	0.000	0.000	A-21
Aux/B[e] Liquid	4.3×10^{-1} (7.0×10^{-2}) μCi/ml	9.5×10^8 (7.0×10^7) ml	3.5×10^8 (6.3×10^7)	0.001	0.000	A-22
Aux/B Gas Release	NM					

Total Core Inventory[a]	=	6.2×10^{11} μCi
Inventory Accounted for to July 1987	=	6.9×10^{11} ($\pm 6.3 \times 10^{10}$) μCi
Fraction Accounted for	=	1.122 (±0.101)

a. All decay data corrected to July 1987.
Half Life = 11030 days
Decay Factor Since Accident = 0.8298

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured.

d. R/B = Reactor Building.

e. Aux/B = Auxiliary Building.

3.2.4 Estimated Inventory of Low Volatility Fission Products (Ruthenium, Cerium, and Europium)

Table 12 summarizes the estimated inventory for Ru-106. Since slightly less than half of the estimated core inventory is accounted for (most of this is associated with the intact rods), additional degraded core material examination data will be necessary to improve the ruthenium inventory estimate.

Table 13 summarizes the estimated Ce-144 inventory. The ORIGEN2 inventory is within the measured limits shown in Table 13. The uncertainty in the measured total core inventory is relatively high due largely to uncertainties associated with the upper core and lower plenum debris data.

Table 14 summarizes the estimated Eu-154 inventory. The estimated total core activity appears to be within approximately 10% of the ORIGEN2 total core value.

TABLE 12. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Ru-106[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (μCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	5.5×10^{2} (5.0×10^{0}) μCi/g	2.4×10^{7} (1.2×10^{6}) g	1.4×10^{9} (7.2×10^{7})	0.110	0.041	A-1
Molten Core Zone	1.5×10^{1} (1.1×10^{0}) μCi/g	2.6×10^{7} (6.6×10^{6}) g	1.6×10^{8} (4.3×10^{7})	0.012	0.014	A-2
Intact Rods	3.6×10^{12} (3.6×10^{10}) μCi	3.3×10^{-1} (2.5×10^{-2})	4.4×10^{9} (3.5×10^{8})	0.344	0.934	A-3
Core Former Region	1.5×10^{1} (1.1×10^{0}) μCi/g	5.0×10^{6} (2.0×10^{6}) g	3.1×10^{7} (1.3×10^{7})	0.002	0.001	A-4
Core Support Assembly	1.5×10^{1} (1.1×10^{0}) μCi/g	5.0×10^{6} (2.0×10^{6}) g	3.1×10^{7} (1.3×10^{7})	0.002	0.001	A-5
Lower Plenum Debris	1.5×10^{1} (1.1×10^{0}) μCi/g	1.5×10^{7} (5.0×10^{6}) g	9.3×10^{7} (3.2×10^{7})	0.007	0.008	A-6
Reactor Cooling System						
Hot Leg surfaces	NM[c]					
Upper Plenum Surfaces	3.1×10^{0} (7.0×10^{-2}) μCi/cm^2	3.5×10^{6} (7.1×10^{5}) cm^2	1.0×10^{6} (2.0×10^{5})	0.000	0.000	A-8
SG Surfaces	3.4×10^{-2} (1.0×10^{-4}) μCi/cm^2	3.7×10^{7} (3.7×10^{6}) cm^2	1.3×10^{5} (1.3×10^{4})	0.000	0.000	A-9
Pressurizer Surfaces	NM					
SG Sediment	NM					
Pressurizer Sediment	4.5×10^{0} ( ) μCi/g	6.6×10^{4} ( ) g	1.1×10^{5} ( )	0.000	0.000	A-12
Demin Sediment	NM					
RCDT Sediment	6.1×10^{1} (3.0×10^{0}) μCi/g	2.6×10^{4} (5.2×10^{3}) g	1.7×10^{5} (3.6×10^{4})	0.000	0.000	A-14
RCS Coolant	NM					
Outside RCS						
R/B[d] Liquid	NM					
R/B Sediment	1.0×10^{2} (7.0×10^{0}) μCi/g	3.8×10^{5} ( ) g	5.8×10^{5} (3.9×10^{4})	0.000	0.000	A-18
R/B Lower Walls	NM					
R/B Upper Surfaces	NM					
R/B Air Space	9.0×10^{-11} (0.00×10^{0}) μCi/cm^3	5.6×10^{10} ( ) cm^3	3.6×10^{-2} ( )	0.000	0.000	A-21
Aux/B[e] Liquid	NM					
Aux/B Gas Release	NM					

a. All decay data corrected to July 1987.
Half Life = 366 days
Decay Factor Since Accident = 0.0036

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured.

d. R/B = Reactor Building.

e. Aux/B = Auxiliary Building.

Total Core Inventory[a]	=	1.3×10^{10} μCi
Inventory Accounted for to July 1987	=	6.2×10^{9} ($\pm 3.6 \times 10^{8}$) μCi
Fraction Accounted for	=	0.478 (±0.028)

TABLE 13. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Ce-144[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (µCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	2.7×10^{3} (7.0×10^{1}) µCi/g	2.4×10^{7} (1.2×10^{6}) g	3.7×10^{9} (2.1×10^{8})	0.222	0.025	A-1
Molten Core Zone	4.2×10^{2} (1.0×10^{0}) µCi/g	2.6×10^{7} (6.6×10^{6}) g	3.6×10^{9} (9.3×10^{8})	0.215	0.488	A-2
Intact Rods	2.4×10^{13} (2.4×10^{11}) µCi	3.3×10^{-1} (2.5×10^{-2})	5.9×10^{9} (4.6×10^{8})	0.350	0.117	A-3
Core Former Region	4.2×10^{2} (1.0×10^{0}) µCi/g	5.0×10^{6} (2.0×10^{6}) g	7.1×10^{8} (2.8×10^{8})	0.042	0.045	A-4
Core Support Assembly	4.2×10^{2} (1.0×10^{0}) µCi/g	5.0×10^{6} (2.0×10^{6}) g	7.1×10^{8} (2.8×10^{8})	0.042	0.045	A-5
Lower Plenum Debris	4.2×10^{2} (1.0×10^{0}) µCi/g	1.5×10^{7} (5.0×10^{6}) g	2.1×10^{9} (7.1×10^{8})	0.126	0.280	A-6
Reactor Cooling System						
Hot Leg surfaces	2.6×10^{-1} (9.0×10^{3}) µCi/cm^2	9.7×10^{5} (9.7×10^{4}) cm^2	1.2×10^{4} (1.2×10^{3})	0.000	0.000	A-7
Upper Plenum Surfaces	7.1×10^{0} (6.8×10^{-1}) µCi/cm^2	3.5×10^{6} (7.1×10^{5}) cm^2	1.2×10^{6} (2.6×10^{5})	0.000	0.000	A-8
SG Surfaces	NM[c]					
Pressurizer Surfaces	NM					
SG Sediment	2.6×10^{1} ( ) µCi/g	8.0×10^{3} (5.0×10^{3}) g	1.1×10^{5} (6.7×10^{4})	0.000	0.000	A-11
Pressurizer Sediment	8.3×10^{0} ( ) µCi/g	6.6×10^{4} ( ) g	1.5×10^{5} ( )	0.000	0.000	A-12
Demin Sediment	NM					
RCBT Sediment	NM					
RCS Coolant	4.9×10^{-2} ( ) µCi/ml	3.3×10^{8} ( ) ml	3.7×10^{4} ( )	0.000	0.000	A-15
Outside RCS						
R/B[d] Liquid	NM					
R/B Sediment	6.6×10^{1} (3.0×10^{0}) µCi/g	3.8×10^{5} ( ) g	1.1×10^{5} (5.0×10^{3})	0.000	0.000	A-18
R/B Lower Walls	NM					
R/B Upper Surfaces	NM					
R/B Air Space	8.0×10^{-11} ( ) µCi/cm^3	5.6×10^{10} ( ) cm^3	7.7×10^{-3} ( )	0.000	0.000	A-21
Aux/B[e] Liquid	NM					
Aux/B Gas Release	NM					

Total Core Inventory[a]	=	1.7×10^{10} µCi
Inventory Accounted for to July 1987	=	1.7×10^{10} ($\pm 1.3 \times 10^{9}$) µCi
Fraction Accounted for	=	0.997 (± 0.079)

a. All decay data corrected to July 1987.
Half Life = 284 days
Decay Factor Since Accident = 0.0007

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured

d. R/B = Reactor Building.

e. Aux/B = Auxiliary Building.

TABLE 14. FISSION PRODUCT DISTRIBUTION AND INVENTORY OF Eu-154[a]

	Activity (±uncertainty)[b]	Associated Vol/Mass/Area (±uncertainty)[b]	Inventory (±uncertainty)[b] (μCi)	Fractional Core Inventory	Fractional Uncertainty Contribution	Data Table
Degraded Core						
Upper Core Debris	4.4×10^{1} (6.0×10^{-1}) μCi/g	2.4×10^{7} (1.2×10^{6}) g	7.9×10^{8} (4.1×10^{7})	0.164	0.016	A-1
Molten Core Zone	3.8×10^{1} (3.0×10^{-1}) μCi/g	2.6×10^{7} (6.6×10^{6}) g	8.7×10^{8} (2.2×10^{8})	0.180	0.479	A-2
Intact Rods	9.6×10^{9} (9.6×10^{7}) μCi	3.3×10^{-1} (2.5×10^{-2})	1.6×10^{9} (1.2×10^{8})	0.327	0.143	A-3
Core Former Region	3.8×10^{1} (3.0×10^{-1}) μCi/g	5.0×10^{6} (2.0×10^{6}) g	1.7×10^{8} (6.8×10^{7})	0.035	0.044	A-4
Core Support Assembly	3.8×10^{1} (3.0×10^{-1}) μCi/g	5.0×10^{6} (2.0×10^{6}) g	1.7×10^{8} (6.8×10^{7})	0.035	0.044	A-5
Lower Plenum Debris	3.8×10^{1} (3.0×10^{-1}) μCi/g	1.5×10^{7} (5.0×10^{6}) g	5.1×10^{8} (1.7×10^{8})	0.106	0.275	A-6
Reactor Cooling System						
Hot Leg surfaces	NM[c]					
Upper Plenum Surfaces	7.1×10^{-3} (3.6×10^{-4}) μCi/cm^2	3.5×10^{6} (7.1×10^{5}) cm^2	1.9×10^{4} (3.9×10^{3})	0.000	0.000	A-8
SG Surfaces	NM					
Pressurizer Surfaces	NM					
SG Sediment	NM					
Pressurizer Sediment	NM					
Demin Sediment	NM					
RCDT Sediment	NM					
RCS Coolant	NM					
Outside RCS						
R/B[d] Liquid	NM					
R/B Sediment	NM					
R/B Lower Walls	NM					
R/B Upper Surfaces	NM					
R/B Air Space	2.0×10^{-11} ( ) μCi/cm^3	5.6×10^{10} ( ) cm^3	6.1×10^{-1} ( )	0.000	0.000	A-21
Aux/B[e] Liquid	NM					
Aux/B Gas Release	NM					

a. All decay data corrected to July 1987.
Half Life = 2993 days
Decay Factor Since Accident = 0.5027

b. Where no uncertainty is shown, it is unknown.

c. NM = Not Measured.

d. R/B = Reactor Building.

e. Aux/B = Auxiliary Building.

Total Core Inventory[a]	=	4.8×10^{9} μCi
Inventory Accounted for to July 1987	=	4.1×10^{9} ($\pm 3.2 \times 10^{8}$) μCi
Fraction Accounted for	=	0.847 (±0.067)

4. SUMMARY AND RECOMMENDATIONS

TMI-2 defueling work has allowed identification of the major fission product repositories. These repositories have been visually inspected to allow estimates to be made of the amount of degraded core material within each. Samples from most repositories have been acquired for characterizing the retained fission products. The currently available data have been compiled for selected isotopes representing the noble gases and the high-, medium-, and low-volatility fission products. Calculations are presented summarizing the total core activity and associated uncertainty (for selected radioisotopes) from all major repositories.

The work presented is based on four important assumptions that limit the accuracy of the results. These include:

1. The specific activity (µCi/g) of the degraded core material in the molten core zone, CFA region, CSA region, and lower plenum region are based on the measured activities of a few lower plenum debris samples.

2. The total core inventory is assumed to be equal to the ORIGEN2 calculated values. No error has yet been associated with the validity of these values. Measurements on intact fuel from known locations will allow the uncertainty in the ORIGEN2 calculated total core inventory to be estimated.

3. It has been assumed that the limited samples from each zone are typical and fully representative of all material in that zone. Additional analysis will be required to estimate the variability in the measured degraded core zone activities as the data become available.

4. Fission product retention in the intact rods was assumed to be 99% (±1) and 95% (±5) for the low-to-medium and high

volatility fission products, respectively. Measured retention data from the intact rods will provide the necessary data to confirm these assumed values.

Table 15 summarizes, for selected isotopes, the fraction of the total core activity estimated from all major repositories. The results indicates the noble gas inventory to be accounted for to within about 10-20% of the initial (ORIGEN2) inventory. However, there appears to be a wide range in accountability for all other fission product groupings, i.e. high-, medium-, and low-volatility groups.

For the highly volatile fission products, cesium appears to be accounted for to within 10%; however, significant iodine (up to 35%) has yet to be accounted for. Additional data evaluation of the early reactor coolant samples will be evaluated to provide an independent check on the iodine inventory transferred to the RCS coolant and provide better agreement between the noble gas and iodine inventories.

Considering the medium-volatile fission products, the fraction of the core inventory of antimony measured to date is less than 50%, while the estimated fractional inventory of strontium appears to agree with the ORIGEN2 total core value.

For the low-volatile fission products, the Ruthenium inventory appears to be low (less than 50%) while the inventory for cerium and europium appear to be within approximately 10% of the ORIGEN2 values.

Since limited activity data is currently available from most degraded core material zones, the estimated inventory values summarized above should be viewed as preliminary and are expected to change somewhat as more core material examination data becomes available. The following recommendations are presented to improve the data and allow evaluation of the above assumptions.

TABLE 15. SUMMARY OF FISSION PRODUCT INVENTORY CALCULATIONS (July 1987)

Fission Product	Current Estimate		Previous Estimate[a] (Ref. 4)
	High Volatility		
Kr-85	86%	(±4%)	47%
Cs-137	104%	(±3%)	50-73%
I-129	68%	(±3%)	
	Medium Volatility		
Sb-125	43%	(±3%)	8-40%
Sr-90	112%	(±10%)	15-63%
	Low Volatility		
Ru-106	48%	(±3%)	--b
Ce-144	100%	(±8%)	26-130%
Eu-154	85%	(±7%)	--b

a. Lower estimate based on limited measured data for only the upper core debris. Upper estimate based on assumption of all core material activity equal to the measured upper plenum debris activity concentration.

b. Not estimated.

1. Examination Data. Additional data are necessary to characterize all regions of the degraded core materials. These include:

   a. Lower Plenum Debris. Grab samples are required of the lower plenum loose debris at all five core inspection locations and at three axial elevations within the original post-accident debris. Samples of the consolidated (lava-like) material are also necessary since there appears to be a significant region of consolidated debris. Samples of all unique debris configurations are also necessary, including the possible nonfuel material adjacent to the reactor vessel. To meet these requirements, removal of the lower plenum debris material must be done carefully.

   b. Core Former Wall Debris. Selected samples are necessary to evaluate the composition and retained fission products from material within the CFA region. Since the mass is estimated to be relatively small compared to other fuel-containing regions, a limited number of samples (5) will provide the data to evaluate variability in the measured activity values vs. location of the material within the region. Also, visual inspection to confirm any significant variation in the CFA debris characteristics (particle size, texture, etc.) and the extent of the CFA volume containing debris material will be necessary.

   c. Core Support Assembly. Selected samples are necessary to evaluate the composition and retained fission products vs. location for the CSA material. Also, visual characterization of the volume of fuel in the CSA is necessary. Because the volume of CSA material is known to be small, only a few (5) samples will be necessary to evaluate sample variability.

2. ORIGEN2 Verification Examinations. The sample examinations on intact fuel pellets from known core locations will provide data

to assess the uncertainty in the ORIGEN2 calculated core activities. As noted above, no uncertainty is assumed for the ORIGEN2 total core activity; however, the uncertainty in these values may be as high as 20-30% for certain isotopes. The overestimation of the measured fractional core inventory for Sr-90 may be due in part to inaccurate ORIGEN2 values for the total core inventory.

3. Data Qualification. Data qualification will be required to verify all data used in the inventory analysis. This includes an extensive evaluation of the Appendix A data included in this report.

The measured fission product data is extensive as evidenced by Tables 7-14. The data used in these tables have been condensed from many data sources. An important end-product of the TMI-2 Accident Evaluation Program is to organize the data necessary to complete the inventory estimates in a fission product data base. The fission product data base should have the following capabilities.

1. Documentation of the data analysis assumptions used to interpret the data and data references for each TMI-2 repository in a manner similar to that documented in Appendix A.

2. The qualified data should be linked interactively to an algorithm to provide summary inventory results similar to the tables presented in Section 3 on all measured fission product isotopes.

3. All repository summary data entered into the fission product data base will require best-estimate values and uncertainty bounds to represent the variability in the available data.

5. REFERENCES

1. E. Tolman, et al., TMI-2 Core Bore Acquisition Summary Report, EGG-TMI-7385 (Rev. 1), February 1987.

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APPENDIX A

SUMMARY OF RADIOLOGICAL CONCENTRATION MEASUREMENTS AND ASSOCIATED CORE MATERIAL FOR THE MAJOR TMI-2 FISSION PRODUCT REPOSITORIES

FIGURES

TABLES

APPENDIX A
SUMMARY OF RADIOLOGICAL CONCENTRATION MEASUREMENTS AND ASSOCIATED CORE MATERIAL FOR THE MAJOR TMI-2 FISSION PRODUCT REPOSITORIES

This Appendix provides the summary data necessary to estimate the overall fission product retention for each of the following regions:

- Degraded core regions
- Reactor cooling system (RCS)
- Ex-RCS regions.

A short description is given for each repository, together with a summary of the data, data references, and major assumptions used in reducing and/or combining the available data for the inventory calculations presented in Section 3.

1. DEGRADED CORE REGION

The configuration of the degraded core has been reasonably characterized from recent defueling data. Figure A-1 summarizes the end-state configuration of the core materials and shows the following unique regions containing significant quantities of fuel material:

- Upper core debris (original core zone),
- Previously molten core material (original core zone),
- Intact rods (original core region),
- Previously molten debris within the core former and baffle plate region, and
- Lower plenum debris.

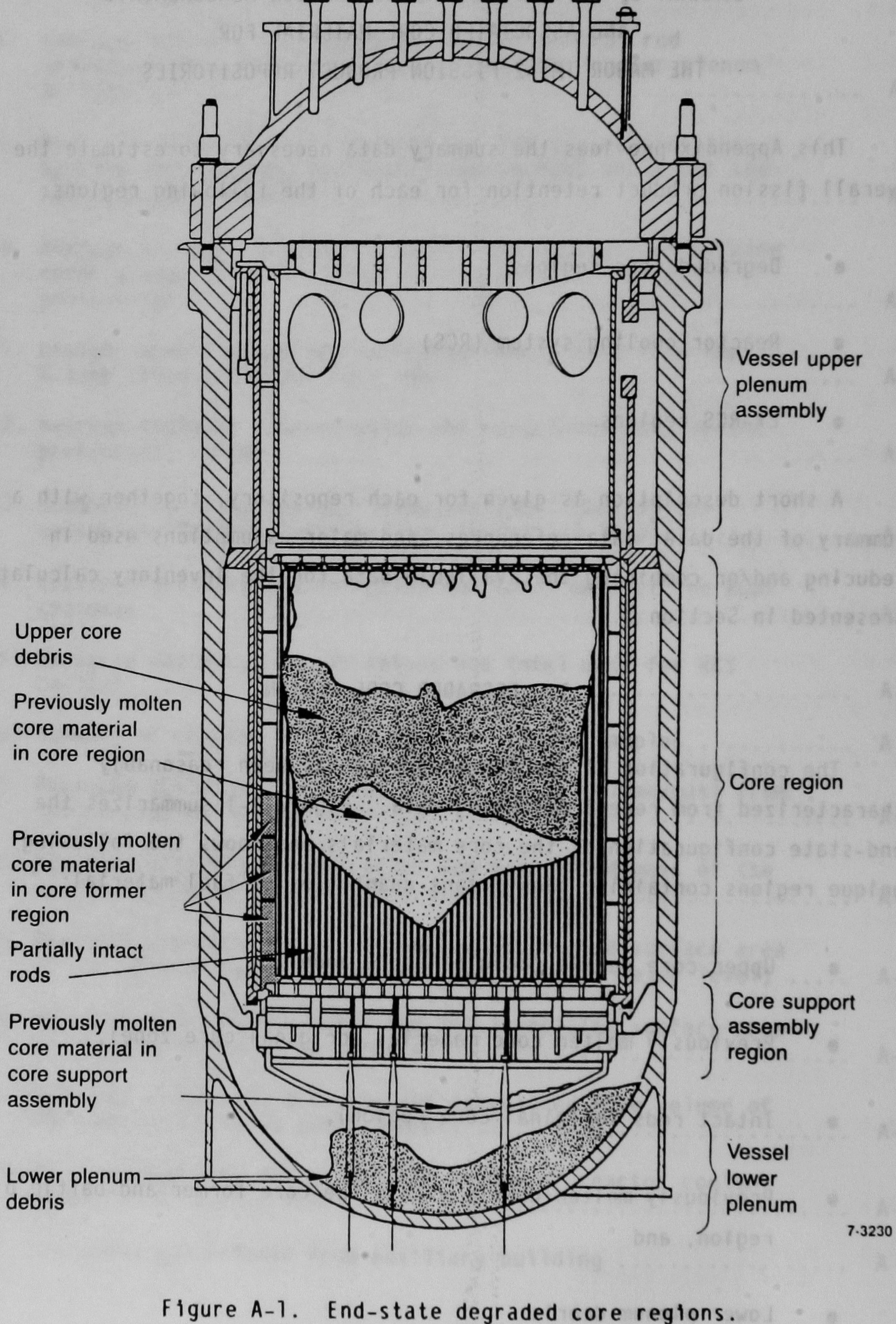

Figure A-1. End-state degraded core regions.

The fission product measurements and associated core mass for each of these regions are discussed in the following subsections.

1.1 Upper Core Debris

A debris bed, ranging from 0.6 to 1.0 m in depth, was supported by a solid crust located at about the core axial mid-plane. The debris bed material was sampled at 11 locations; the debris particles were then examined in the laboratory. Most of the particles examined contained small amounts of previously molten U-Zr-O or U/O, indicating localized peak temperatures of the previously molten material were in the range of 2800-3100 K. However, based on the appearance of most of the debris, it is estimated that the average bulk debris temperature remained below about 2200 K. Details of the examination results are reported in Ref. A-1.

The contour of the debris upper surface was determined using acoustic topography and is known to within a few percent. The debris material was extensively probed to determine the depth to the hard supporting surface. The location of this surface is known accurately to within 1 to 2 inches. Using these data, the volume of the upper debris bed is estimated to be 6.7 m^3. Using the measured bulk debris density of 4.48 g/cm^3 (Ref. A-1), the upper debris bed mass is estimated to be approximately 30,000 kg. Assuming the uncertainty in the estimated volume to be 5% and the uncertainty in bulk density to be $\pm 1.0\ g/cm^3$ (from the measured data), the uncertainty in the upper debris mass is 6700 kg, or about 22%.

GPU Nuclear defueling records estimate the upper core debris mass removed to be 23,799 kg (Ref. A-2). The measured defueling mass is estimated to be within 5%. Since the defueling data is thought to be more precise than estimates based on the measured volume and density, the defueling data will be used for the inventory calculation.

The first six upper core debris samples were obtained during September and October, 1983. Five additional samples were acquired in March 1984.

The locations of the eleven samples in the TMI-2 core are shown in Fig. A-2. The eleven samples were shipped to INEL. Ten samples were retained at INEL and one sample was shipped to B&W for examination. Six additional upper core debris samples were taken on April 12, 1986.

Average radionuclide concentrations of the combined upper core debris bulk samples are listed in Tables 21 and 22 of Ref. A-1. Table A-1 shows the results of sample weighted radionuclide concentrations obtained from these tables and the estimated end-state mass of the upper core debris (from Ref. A-3).

1.2 Previously Molten Core Material

The molten core zone is a crucible-shaped region with a maximum depth of approximately 1.2 m near the center, decreasing to 15-30 cm near the periphery of the molten zone. Two regions containing previously molten material were observed,

1. a region of previously molten material surrounding damaged but intact fuel pellets near the periphery of the molten core zone, and

2. a region of uniformly molten material in the central regions of the molten zone; no evidence of intact fuel pellets or rod structures were observed in this region.

Samples from each of these regions are being examined in the laboratory to characterize composition and isotopic levels. Final data are expected by the end of calendar year 1987.

The lower interface between the molten core zone and the lower intact rod stubs has been estimated by contour mapping from the core bore inspection data. The best-estimate volume of the molten core region based on the core bore inspection data is $3.65 \pm 0.91\ m^3$ ($129 \pm 32\ ft^3$) (Ref. A-4). The mass of the previously molten core region is estimated to

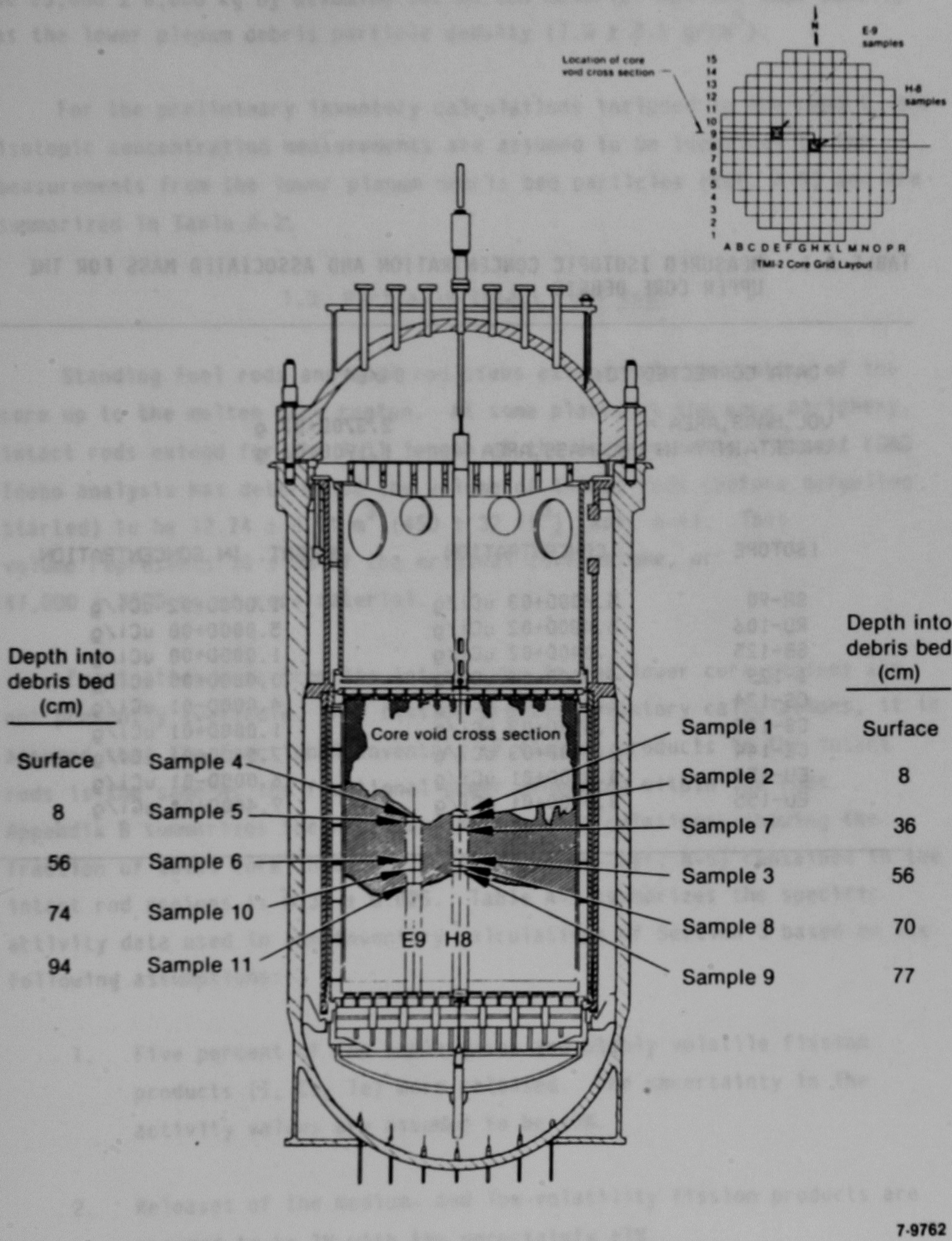

Figure A-2. Location of upper core debris grab samples.

TABLE A-1. MEASURED ISOTOPIC CONCENTRATION AND ASSOCIATED MASS FOR THE UPPER CORE DEBRIS

DATA CORRECTED TO: 04/84

VOL,MASS,AREA = 2.370D+07 g
UNCERTAINTY IN VOL,MASS,AREA = 1.190D+06 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	5.600D+03 uCi/g	1.000D+02 uCi/g
RU-106	5.480D+02 uCi/g	5.000D+00 uCi/g
SB-125	1.040D+02 uCi/g	1.000D+00 uCi/g
I-129	4.380D-04 uCi/g	0.000D+00 uCi/g
CS-134	6.640D+01 uCi/g	4.000D-01 uCi/g
CS-137	1.550D+03 uCi/g	1.000D+01 uCi/g
CE-144	2.740D+03 uCi/g	7.000D+01 uCi/g
EU-154	4.350D+01 uCi/g	6.000D-01 uCi/g
EU-155	8.990D+01 uCi/g	9.400D+00 uCi/g

be 25,600 ± 6,600 kg by assuming the molten material has the same density as the lower plenum debris particle density (7.0 ± 0.5 g/cm^3).

For the preliminary inventory calculations included in the report, the isotopic concentration measurements are assumed to be identical to the measurements from the lower plenum debris bed particles (Ref. A-5) and are summarized in Table A-2.

1.3 Partially Intact Fuel Rods

Standing fuel rods and fuel rod stubs extend from the bottom of the core up to the molten core region. At some places at the core periphery, intact rods extend for the full length of the fuel assembly. Recent EG&G Idaho analysis has determined the volume of intact rods (before defueling started) to be 12.74 ± 0.91 m^3 (450 ± 32 ft^3) (Ref. A-4). This volume represents 38 ± 3% of the original core volume, or 47,000 ± 3800 kg of core material.

Examination data from the intact rods in the lower core regions are not presently available. For fission product inventory calculations, it is assumed that the fractional inventory of fission products in the intact rods is the same as the fractional power generated within the rods. Appendix B summarizes the necessary data and calculations, showing the fraction of total core inventory (ORIGEN2 value--Ref. A-6) contained in the intact rod regions is 0.32 ± 0.025. Table A-3 summarizes the specific activity data used in the inventory calculations of Section 3 based on the following assumptions:

1. Five percent of the noble gases and highly volatile fission products (I, Cs, Te) were released. The uncertainty in the activity values are assumed to be ±5%.

2. Releases of the medium- and low-volatility fission products are assumed to be 1% with the uncertainty ±1%.

TABLE A-2. ESTIMATED ISOTOPIC CONCENTRATION AND ASSOCIATED MASS FOR MOLTEN CORE ZONE

DATA CORRECTED TO: 04/86

VOL,MASS,AREA = 2.560D+07 g
UNCERTAINTY IN VOL,MASS,AREA = 6.600D+06 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	7.070D+03 uCi/g	1.700D+02 uCi/g
RU-106	1.460D+01 uCi/g	1.100D+00 uCi/g
SB-125	7.950D+00 uCi/g	1.300D+00 uCi/g
I-129	5.600D-05 uCi/g	3.200D-06 uCi/g
CS-134	2.680D+01 uCi/g	6.000D-01 uCi/g
CS-137	8.720D+02 uCi/g	2.000D+00 uCi/g
CE-144	4.230D+02 uCi/g	1.000D+00 uCi/g
EU-154	3.750D+01 uCi/g	3.000D-01 uCi/g
EU-155	0.000D+00 uCi/g	0.000D+00 uCi/g

TABLE A-3. ESTIMATED FISSION PRODUCT INVENTORY IN INTACT RODS

Isotope	Estimated Inventory in Intact Rods[a]	Uncertainty in Estimated Inventory
SR-90	7.460D+11 uCi	7.460D+09 uCi
RU-106	3.580D+12 uCi	3.580D+10 uCi
SB-125	1.220D+11 uCi	1.220D+09 uCi
I-129	2.054D+05 uCi	1.080D+04 uCi
CS-134	1.891D+11 uCi	9.950D+09 uCi
CS-137	8.113D+11 uCi	4.270D+10 uCi
CE-144	2.360D+13 uCi	2.360D+11 uCi
EU-154	9.550D+09 uCi	9.550D+07 uCi
EU-155	3.240D+10 uCi	3.240D+08 uCi
KR-85	9.206D+10 uCi	4.850D+09 uCi

a. See Appendix B for assumptions and data for calculating inventory.

1.4 Previously Molten Core Material in the Core Former Zone

Recent defueling data has confirmed the existence of significant amounts of previously molten core debris in the core former/baffle plate regions at the periphery of the core. Reference A-7 suggests that an upper bound estimate of core debris in the core former region is 6200 kg (13,600 lb). However, based on uncertainties in the debris density and the assumption that the region is filled below the upper surface of the debris, a best-estimate mass of degraded core material for this region is assumed to be 5000 ± 2000 kg.

No CFA samples have yet been received and examined; however, based on the best-estimate accident scenario (Ref. A-8), the previously molten material is thought to be part of the major core material relocation at 224 minutes. Thus, the composition and fission products are expected to be similar to that of the lower plenum debris.

For the inventory calculations presented in this report, the isotopic concentrations are assumed to be identical to the lower plenum debris measurements (Ref. A.5). These values, together with the estimated mass of core material in the CFA, are presented in Table A-4.

1.5 Previously Molten Core Material in the Core Support Assembly (CSA) Region

The fuel debris in the CSA region (see Fig. A-1) is at present not well quantified. During the core bore inspection of the CSA region, previously molten material was observed only in the east quadrant of the vessel and in these regions only limited quantities were observable. In addition, video inspection of the lower plenum regions has shown previously molten, lava-like material to rest on the upper surfaces of the elliptical flow distributor plate. Based on the limited inspection data, the mass of the previously molten core material in the CSA region is estimated to be 5000 ± 2000 kg. As defueling progresses, additional CSA inspection data

TABLE A-4. ESTIMATED ISOTOPIC CONCENTRATION AND ASSOCIATED MASS FOR DEBRIS IN THE CORE FORMER REGION

DATA CORRECTED TO: 04/86

VOL,MASS,AREA = 5.0000D+06 g
UNCERTAINTY IN VOL,MASS,AREA = 2.000D+06 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	7.070D+03 uCi/g	1.700D+02 uCi/g
RU-106	1.460D+01 uCi/g	1.100D+00 uCi/g
SB-125	7.950D+00 uCi/g	1.300D+00 uCi/g
I-129	5.600D-05 uCi/g	3.200D-06 uCi/g
CS-134	2.680D+01 uCi/g	6.000D-01 uCi/g
CS-137	8.720D+02 uCi/g	2.000D+00 uCi/g
CE-144	4.230D+02 uCi/g	1.000D+00 uCi/g
EU-154	3.750D+01 uCi/g	3.000D-01 uCi/g
EU-155	0.000D+00 uCi/g	0.000D+00 uCi/g

will allow more accurate characterization of the material within these regions. At present, no examination data exists from core material in the CSA regions.

No CSA samples have yet been received and examined; however, based on the best-estimate accident scenario (Ref. A-8), the previously molten material is thought to be part of the major core material relocation at 224 min. Thus, the composition and fission products are expected to be similar to that of the lower plenum debris.

For the inventory calculations presented in this report, the isotopic concentrations are assumed to be identical to the lower plenum debris measurements (Ref. A.5). These values, together with the estimated mass of core material in the CSA, are presented in Table A-5.

1.6 Lower Plenum Debris

The original post-accident lower plenum debris has been characterized via five separate video inspections. Note that since the original inspections, an estimated five tons of fine debris has relocated to the lower plenum region as a result of drilling through the upper core material. The debris bed configuration, based on these data, is summarized in Ref. A-9.

The lower plenum debris material consists of a wide range of material shapes, sizes, and textures. In the plenum north quadrant region, the debris has the appearance of a large, lava-like cliff which abuts the outermost row of instrument guide tubes, while the region in front of the rubble cliff has almost no debris. The cliff appears to be solid, with a smooth surface interlaced with cracks and some large chunks which appear to be loose. The rubble near the west quadrant has a large number of 2- to 5-cm diam. irregular-shaped pieces interspersed with much finer material. The material in the east quadrant appears to have the largest particles, some having diameters (although irregular in shape) ranging from 7 to 20 cm. The larger pieces appear to be porous, with small cracks and smooth surfaces. In the south quadrant, the debris bed looks more uniform and

TABLE A-5. ESTIMATED ACTIVITY AND ASSOCIATED MASS FOR CSA DEBRIS

DATA CORRECTED TO: 04/86

VOL,MASS,AREA = 5.000D+06 g
UNCERTAINTY IN VOL,MASS,AREA = 2.000D+06 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	7.070D+03 uCi/g	1.700D+02 uCi/g
RU-106	1.460D+01 uCi/g	1.100D+00 uCi/g
SB-125	7.950D+00 uCi/g	1.300D+00 uCi/g
I-129	5.600D-05 uCi/g	3.200D-06 uCi/g
CS-134	2.680D+01 uCi/g	6.000D-01 uCi/g
CS-137	8.720D+02 uCi/g	2.000D+00 uCi/g
CE-144	4.230D+02 uCi/g	1.000D+00 uCi/g
EU-154	3.750D+01 uCi/g	3.000D-01 uCi/g

appears to be a transition region between the larger pieces in the east quadrant and the smaller debris towards the center. The debris in this region was easily dislodged by the camera and lights. The debris bed at the core bore inspection locations, towards the center of the vessel, was uniform and smooth with only a few larger pieces in the range of 1 to 2.5 cm.

The volume and mass of the material in the lower plenum has been estimated at 15,000 ± 5,000 kg (before drilling of the upper core region). Sixteen debris samples were obtained from regions in the south and west quadrants of the vessel. These samples were examined in the laboratory for activity and material composition and the results are contained in Ref. A-5. More samples will be retrieved and examined, including large bulk samples of the fine debris, pieces of the consolidated lava-like material, and material adjacent to the vessel walls. In addition, more inspection data will be obtained characterizing the debris distribution as defueling progresses. Thus, the data presented here should be viewed as preliminary with fairly large uncertainties relative to typicality of the overall lower plenum debris material.

Sample examination results are listed in Table F.1 to Table F.9 of Ref. A-5. Table A-6 summarizes the fission product concentrations and associated mass for the lower plenum debris used for the inventory calculations of Section 3. The concentration data were obtained by averaging the data from Table F.1 to Table F.9 of Ref. A-5.

TABLE A-6. MEASURED ISOTOPIC CONCENTRATION AND ASSOCIATED MASS FOR LOWER PLENUM DEBRIS

DATA CORRECTED TO: 04/86

VOL,MASS,AREA = 1.5000D+07 g
UNCERTAINTY IN VOL,MASS,AREA = 5.0000D+06 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	7.0700D+03 uCi/g	1.7000D+02 uCi/g
RU-106	1.4600D+01 uCi/g	1.1000D+00 uCi/g
SB-125	7.9500D+00 uCi/g	1.3000D+00 uCi/g
I-129	5.6000D-05 uCi/g	3.2000D-06 uCi/g
CS-134	2.6800D+01 uCi/g	6.0000D-01 uCi/g
CS-137	8.7200D+02 uCi/g	2.0000D+00 uCi/g
CE-144	4.2300D+02 uCi/g	1.0000D+00 uCi/g
EU-154	3.7500D+01 uCi/g	3.0000D-01 uCi/g
EU-155	0.0000D+00 uCi/g	0.0000D+00 uCi/g

2. REACTOR COOLING SYSTEM

Inspection and examination data are available to estimate the fission product retention for the following regions of the primary cooling system or major RCS flow paths to the containment building:

- Hot leg piping surface,
- Upper plenum surfaces,
- Steam generator surfaces,
- Pressurizer surfaces,
- Steam generator sediment,
- Pressurizer sediment,
- Makeup and purification demineralizer sediment,
- Reactor coolant drain tank sediment, and
- RCS water.

The data relative to each of these sources are discussed in the following subsections.

2.1 Hot Leg Piping Surfaces

A dual-element resistance thermal detector (RTD) was removed from the A-loop hot leg (see Fig. A-3 for location). The RTD tip configuration relative to the hot leg wall is shown in Fig. A-4. The RTD was examined in the laboratory and the results are currently being documented. The measured surface radioactivity assumed for the inventory calculations in Section 3 was taken from Table 6 of Ref. A-10. The total measured surface concentration is obtained by summing up the quantities of each radionuclide

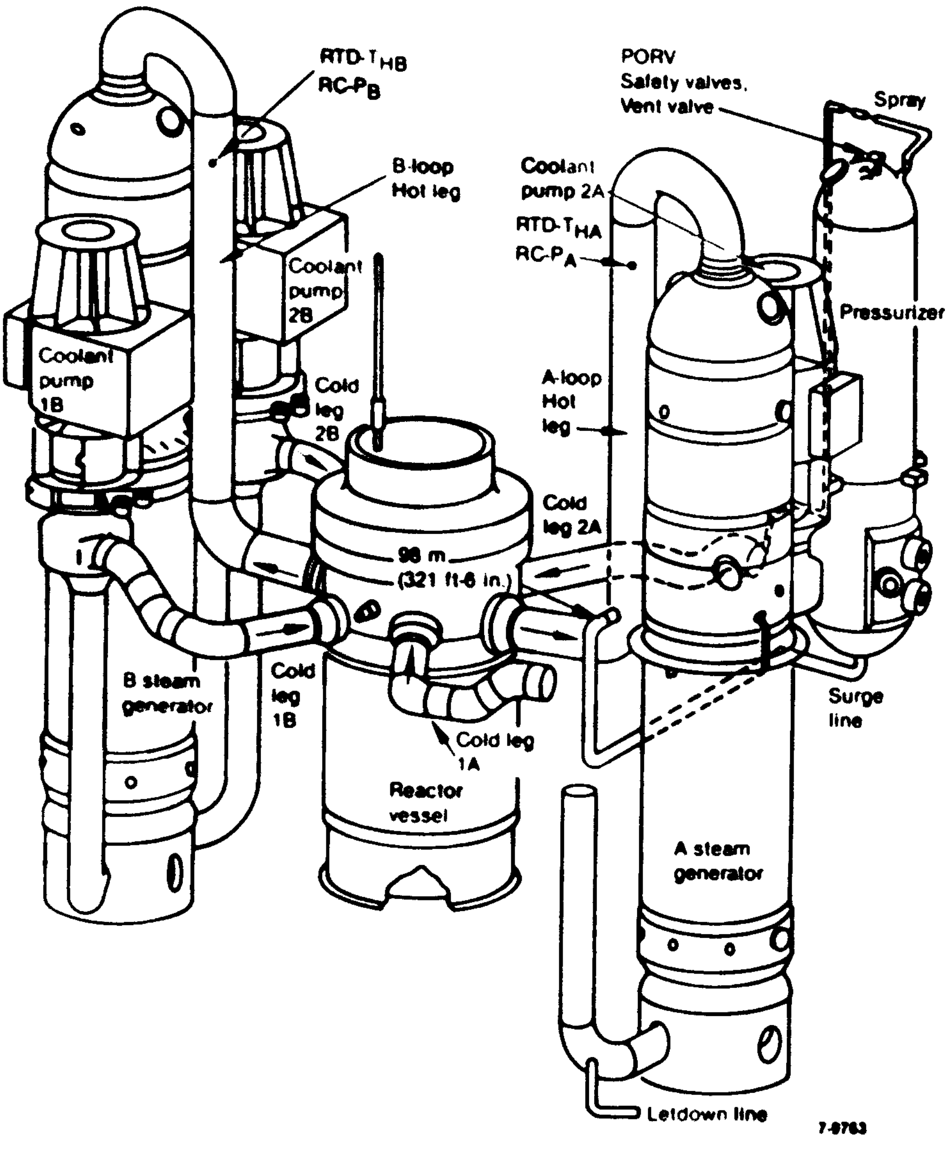

Figure A-3. RCS configuration showing location of A-loop RDT.

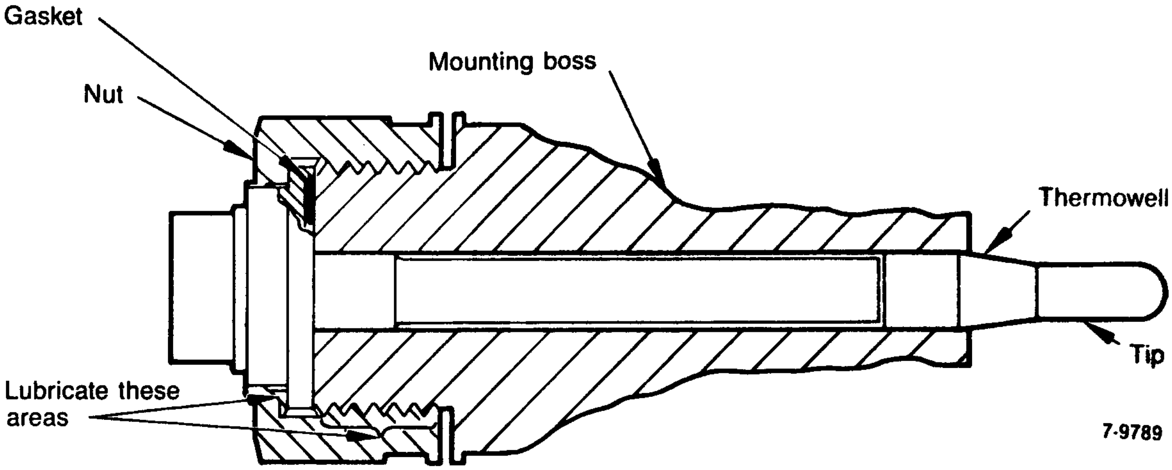

Figure A-4. A-loop RDT tip configuration.

removed during the individual decontamination steps and dividing them by the tip surface area (15.6 cm^2). These data are shown in Table A-7. The estimated surface area of the hot legs assumed to be associated with the RTD data are also summarized in Table A-7 and were taken from Ref. A-11.

No surface samples from the B-loop hot leg surface have been acquired. However, it should be noted that the activity measurements external to the RCS (based on gamma measurements of the steam generator manway cover) indicate that the B-loop surfaces are several times more radioactive than locations on the A-loop (Ref. A-12).

2.2 Upper Plenum Surfaces

Three control rod drive leadscrews were removed from the reactor head as part of the July 1982 remote television inspection of the damaged core. The former locations of the leadscrews removed are shown in Fig. A-5.

Examination of the leadscrews showed two distinct surface deposition regions: (a) an outer layer of easily removed material, and (b) an adherent layer adjacent to the stainless steel. Radiochemical analysis of both the outer and inner deposits were completed. The examination results are presented in Tables 23, 25, 38, and 39 of Ref. A-13 for the brushoff debris and in Tables 26, 28, 40, and 41 of the same document for the decontamination solution.

The radionuclide concentrations used for the inventory calculations were obtained by summing the measured activity from both deposition layers and dividing by the leadscrew area. These data are summarized in Table A-8.

The total upper plenum surface area associated with the average leadscrew concentration was taken from Ref. A-11 (Table C-11) and is also shown in Table A-8.

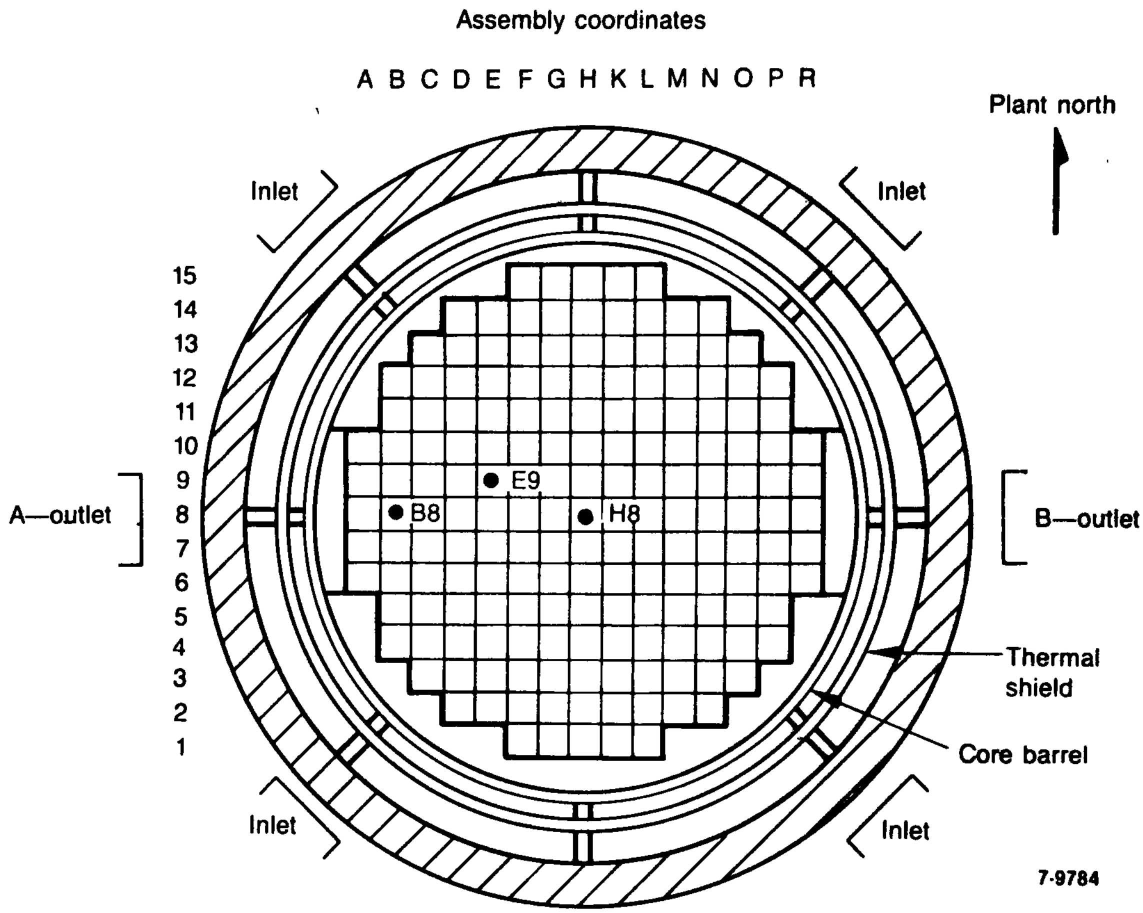

Figure A-5. Location of the control rod leadscrews removed for examination.

TABLE A-7. MEASURED ISOTOPIC DEPOSITION AND ESTIMATED SURFACE AREA FOR RCS HOT LEG SURFACES

DATA CORRECTED TO: 01/84

VOL,MASS,AREA = 9.700D+05 cm2
UNCERTAINTY IN VOL,MASS,AREA = 9.700D+04 cm2

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	9.440D+00 uCi/cm2	2.000D-01 uCi/cm2
SB-125	1.320D-01 uCi/cm2	1.500D-02 uCi/cm2
CS-134	9.410D-01 uCi/cm2	8.000D-03 uCi/cm2
CS-137	2.000D+01 uCi/cm2	1.000D-01 uCi/cm2
CE-144	2.620D-01 uCi/cm2	9.000D-03 uCi/cm2

TABLE A-8. AVERAGE ISOTOPIC ACTIVITY/AREA FOR THE CONTROL ROD LEADSCREWS AND ASSOCIATED AREA OF THE TMI-2 UPPER PLENUM SURFACES

DATA CORRECTED TO: 03/84

VOL,MASS,AREA = 3.540D+06 cm2
UNCERTAINTY IN VOL,MASS,AREA = 7.100D+05 cm2

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	1.980D+01 uCi/cm2	2.000D-01 uCi/cm2
RU-106	3.070D+00 uCi/cm2	7.000D-02 uCi/cm2
SB-125	8.920D+00 uCi/cm2	1.850D+00 uCi/cm2
I-129	5.350D-05 uCi/cm2	3.100D-06 uCi/cm2
TE	5.690D+00 uCi/cm2	0.000D+00 uCi/cm2
CS-134	9.040D+00 uCi/cm2	1.300D-01 uCi/cm2
CS-137	6.570D+01 uCi/cm2	6.500D+00 uCi/cm2
CE-144	7.100D+00 uCi/cm2	6.800D-01 uCi/cm2
EU-154	7.120D-03 uCi/cm2	3.600D-04 uCi/cm2
EU-155	2.450D-02 uCi/cm2	2.000D-04 uCi/cm2

2.3 Steam Generator Surfaces

The manway cover backing (MCB) plates from steam generators A and B (see Fig. A-6) were examined at the Battelle Hot Cell facility in Ohio. The plates were sectioned to obtain samples for radiochemical examination.

The radionuclide activity from the inner surface of each sectioned coupon is listed in Table 3 of Ref. A-14. The effective activity/unit of steam generator surface area was obtained by averaging the measured activity over the surface area associated with each sample. These surface concentration data are shown in Table A-9. The total surface area of the steam generators associated with the surface activity concentrations is also shown in Table A-9 and was taken from Ref. A-14.

2.4 Pressurizer Surfaces

The MCB plate from the pressurizer (see Fig. A-7) was sectioned and examined at the Battelle Hot Cell Facility. The measured radionuclide activity from the inner surface of each sectioned coupon is listed in Table 3 of Ref. A-14. The effective radionuclide surface concentration of the pressurizer MCB plate was obtained by averaging the measurements from each coupon and dividing by the effective surface area of the sample. These data are summarized in Table A-10. The associated total surface area of the pressurizer (from Ref. A-15) is also shown in Table A-10.

2.5 Steam Generator Sediment

The 'tube sheet' of the B-loop steam generator is covered with an estimated 1 to 4 liters of debris with particles ranging in size up to 0.6 cm. A debris sample was vacuumed from the tube sheet on August 30, 1986. The volume of the sample is about 7.7 cm^3. Assuming this material has a density of 3.6 g/cm^3 (the lower average bulk density of the core debris samples from Ref. A-1), the total sample weight is approximately 28 g.

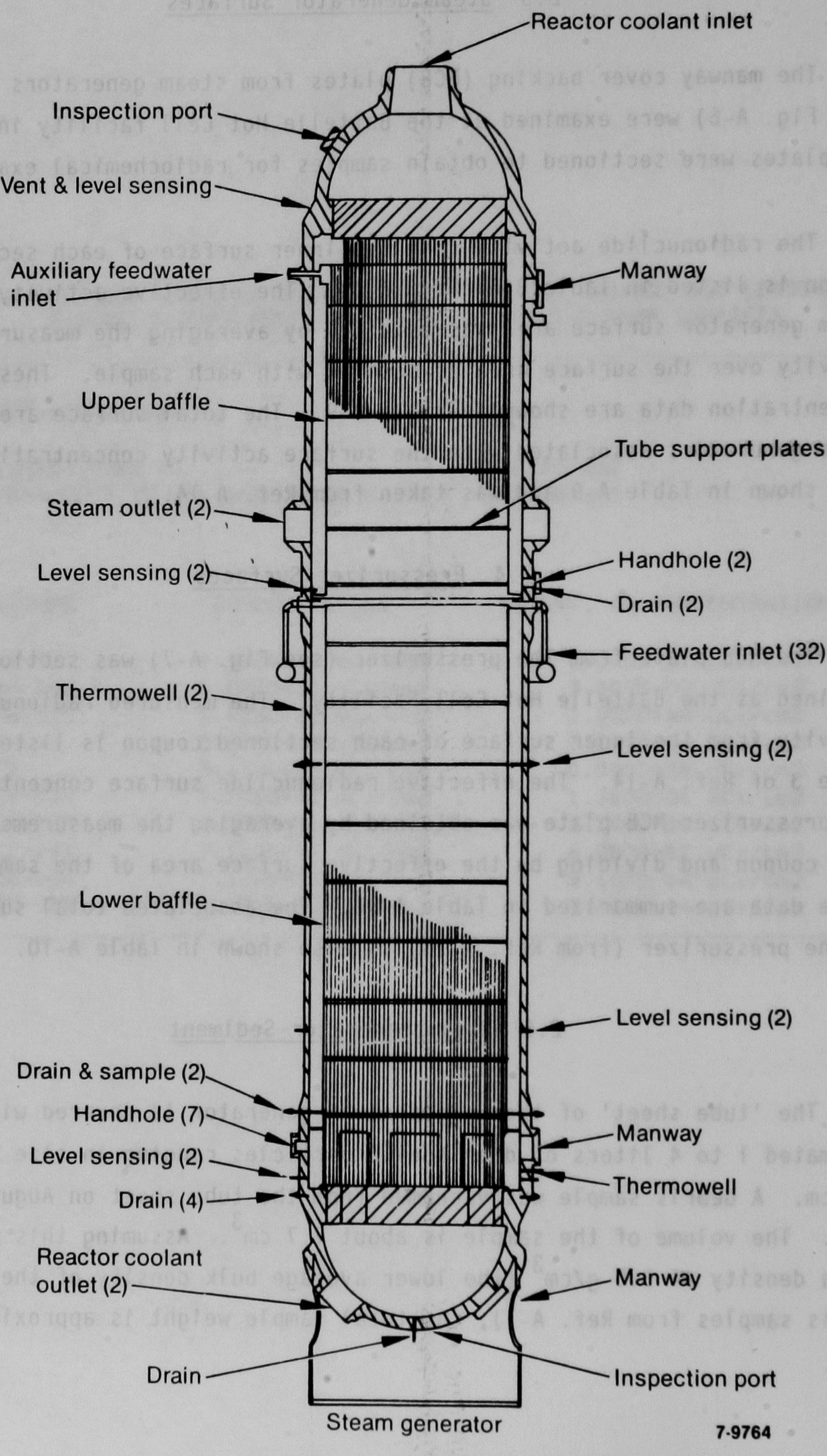

Figure A-6. TMI-2 steam generator configuration.

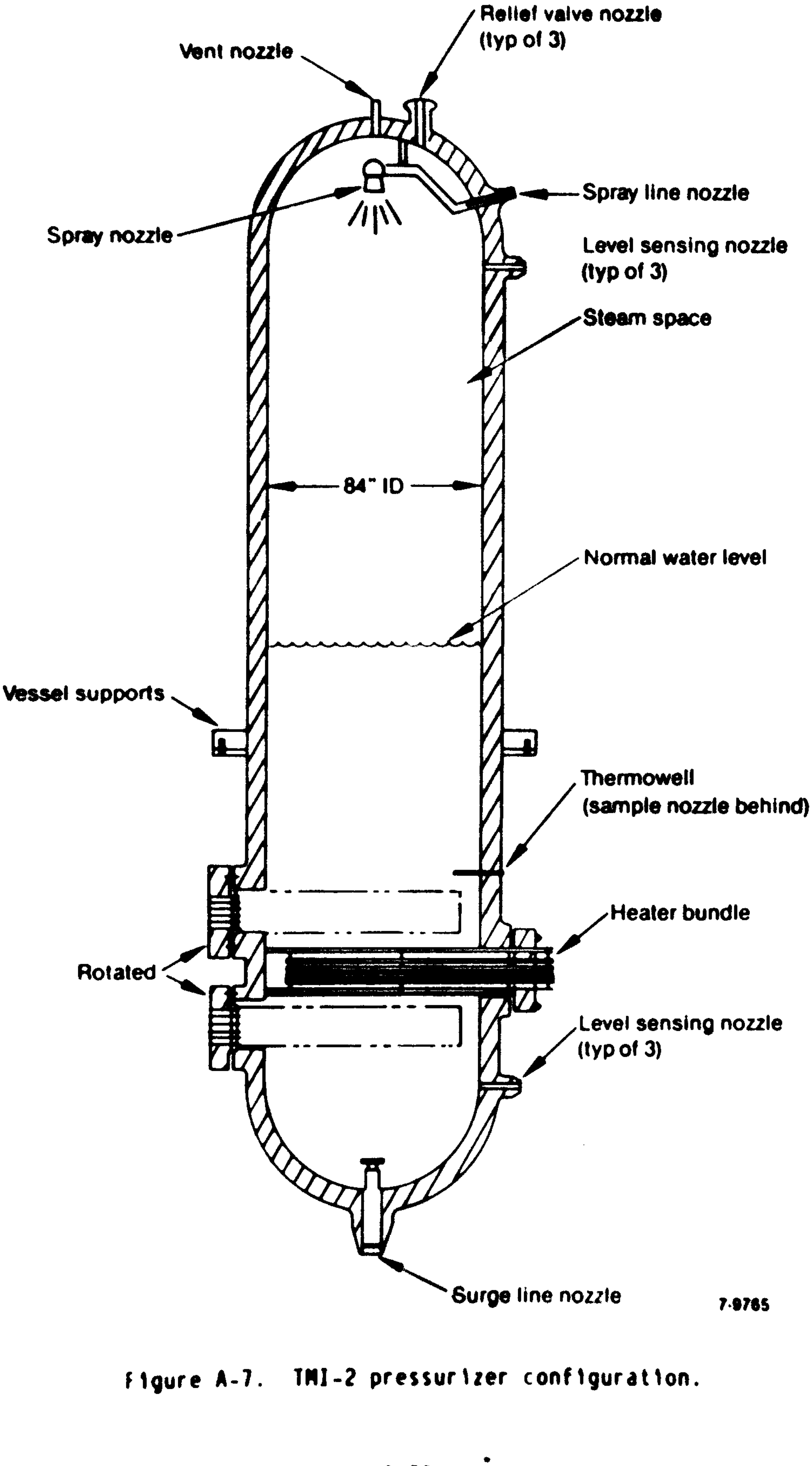

Figure A-7. TMI-2 pressurizer configuration.

TABLE A-9. AVERAGED ISOTOPIC ACTIVITY/AREA FOR THE STEAM GENERATOR BACKING COVER PLATES AND ASSOCIATED SURFACE AREAS FOR THE TMI-2 STEAM GENERATORS

DATA CORRECTED TO: 03/84

VOL,MASS,AREA = 3.700D+07 cm2
UNCERTAINTY IN VOL,MASS,AREA = 3.700D+06 cm2

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	1.260D-01 uCi/cm2	9.000D-03 uCi/cm2
RU-106	3.430D-02 uCi/cm2	1.000D-04 uCi/cm2
SB-125	1.790D-01 uCi/cm2	1.000D-03 uCi/cm2
I-129	2.860D-06 uCi/cm2	2.000D-08 uCi/cm2
CS-134	8.580D-02 uCi/cm2	8.000D-04 uCi/cm2
CS-137	3.640D+00 uCi/cm2	1.000D-02 uCi/cm2

TABLE A-10. AVERAGE ISOTOPIC ACTIVITY/AREA FOR THE PRESSURIZER BACKING COVER PLATES AND ASSOCIATED SURFACE AREA FOR THE TMI-2 PRESSURIZER

DATA CORRECTED TO: 03/84

VOL,MASS,AREA = 1.000D+06 cm2
UNCERTAINTY IN VOL,MASS,AREA = 1.000D+05 cm2

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	3.600D-01 uCi/cm2	1.900D-02 uCi/cm2
SB-125	8.880D-04 uCi/cm2	1.280D-04 uCi/cm2
CS-134	1.180D-03 uCi/cm2	5.000D-05 uCi/cm2
CS-137	4.240D-02 uCi/cm2	3.000D-04 uCi/cm2

The activity of the sample was measured by GPU and the results documented in Ref. A-16. The radionuclide activity/g of debris material was obtained by dividing the measured activity by the sample weight (28 g) and is summarized in Table A-10. Assuming the above density of the debris material, the total debris mass associated with this specific activity is estimated to be between 3.6-14.4 kg. Because of the large uncertainty in the estimated mass, the nominal debris mass and associated uncertainty is assumed to be 8 kg and 5 kg respectively, as noted in Table A-11.

2.6 Pressurizer Sediment

A remote television inspection of the pressurizer internals conducted in December 1985 indicated the presence of sediment within the pressurizer, mostly deposited on the bottom of the pressurizer. The bottom sediment appears to be deepest near the injection nozzle, and decreases toward the periphery. Some larger debris material was also observed. Most of the sediment is comprised of fine particles, easily levitated by localized water disturbance.

A sample of sediment was acquired for examination in December 1985. After initial gamma scans were completed, the sample was divided into liquid and solid components for additional gamma analysis. The measured activities are documented in Table 1 of Ref. A-17. An average radionuclide concentration for the pressurizer sediment was obtained by summing the activity of the solid debris particles greater than 0.45 micron in diameter and dividing by the sample solid weight. This average sediment data is summarized in Table A-12.

A maximum sediment volume of 12 liters was estimated from visual observations (Ref. A-18). A total sediment mass of 66 kg is estimated, based on the volume of the debris and an assumed density of 5.5 g/cm^3 as shown in Table A-12.

TABLE A-11. CESIUM CONCENTRATION AND ESTIMATED DEBRIS MASS FROM THE B-LOOP STEAM GENERATOR TUBE SHEET

DATA CORRECTED TO: 10/86

VOL,MASS,AREA = 8.0000D+03 g
UNCERTAINTY IN VOL,MASS,AREA = 5.0000D+03 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
CS-134	5.730D+01 uCi/g	0.000D+00 uCi/g
CS-137	2.610D+03 uCi/g	0.000D+00 uCi/g
CE-144	2.600D+01 uCi/g	0.000D+00 uCi/g

TABLE A-12. AVERAGE ISOTOPIC CONCENTRATION AND ASSOCIATED MASS OF THE PRESSURIZER SEDIMENT

DATA CORRECTED TO: 01/86

VOL,MASS,AREA = 6.600D+04 g
UNCERTAINTY IN VOL,MASS,AREA = 0.000D+00 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
RU-106	4.500D+00 uCi/g	0.000D+00 uCi/g
SB-125	5.080D-01 uCi/g	0.000D+00 uCi/g
CS-134	1.190D+00 uCi/g	0.000D+00 uCi/g
CS-137	3.720D+01 uCi/g	0.000D+00 uCi/g
CE-144	8.250D+00 uCi/g	0.000D+00 uCi/g
EU-155	1.140D+00 uCi/g	0.000D+00 uCi/g

2.7 Makeup and Purification Demineralizer Sediment

During normal operation, the makeup and purification system receives reactor coolant from the steam generator cold leg for filtration and demineralization. The demineralizer vessels are located in the auxiliary building as shown in Fig. A-8.

GPU Nuclear has estimated that during the first 16.5 hours after initiation of the accident, the makeup and purification system processed about 1.7×10^5 kg (46,000 gal) of water from the RCS and was severely contaminated with fission product radionuclides. The demineralizer resin beds were significantly degraded, both radiolytically and thermally; comparison of the post-accident resin-bed volumes with that of preaccident volumes shows that severe shrinking (~55%) of the resin beds occurred.

The demineralizers were inspected and sampled by GPUN in early 1983 and it was observed that demineralizer "A" contained only dry caked resin, whereas liquid was still present in demineralizer "B". Radionuclide analyses of the resin and liquid phases of the demineralizer "A" and "B" samples are given in Table 2 of Ref. A-19 and are summarized in Table A-13.

Nondestructive assays were employed to estimate the sediment quantity in each demineralizer. Estimated demineralizer vessel loadings are shown in Table 1 of Ref. A-19. These values were used for the associated mass of demineralizer resins for inventory calculations in Section 3.

2.8 Reactor Coolant Drain Tank (RCDT)

The reactor coolant drain tank (RCDT) receives water from the pressurizer when the PORV releases pressure in the reactor system. The RCDT is located in the reactor building basement as shown in Fig. A-9. During the first three days following the accident, an estimated 1.0×10^9 ml of primary coolant escaped from the RCS through the pressurizer to the RCDT (Ref. A-20). Major reactor coolant flow through the RCDT for the first 15 hours of the accident is shown in Fig. A-10 (from Ref. A-21). Leakage through the RCDT continued until July 1982, when the

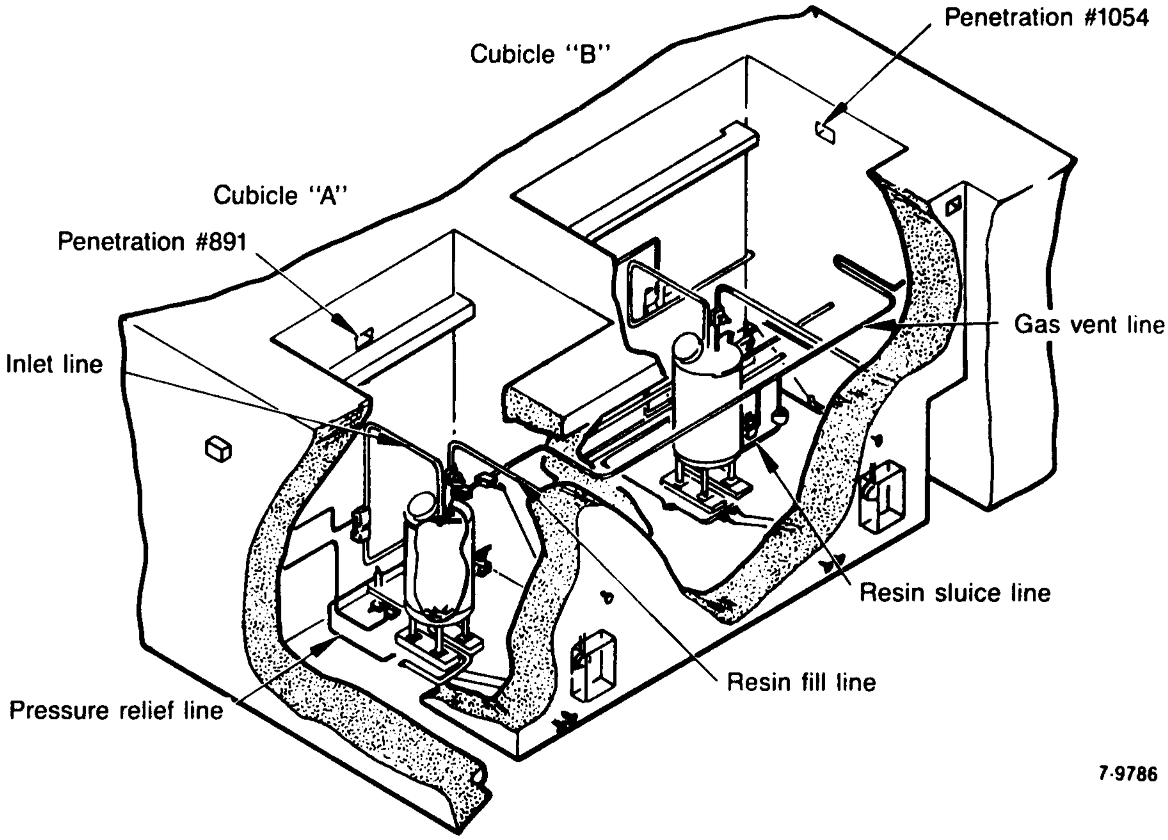

Figure A-8. TMI-2 letdown and makeup demineralizer configuration.

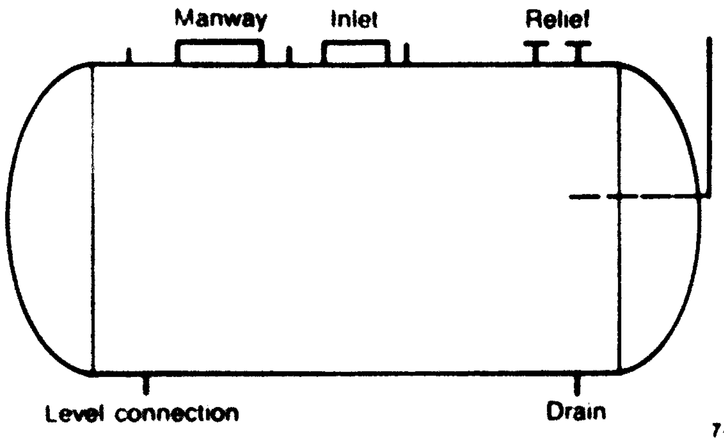

Figure A-9. RCDT configuration.

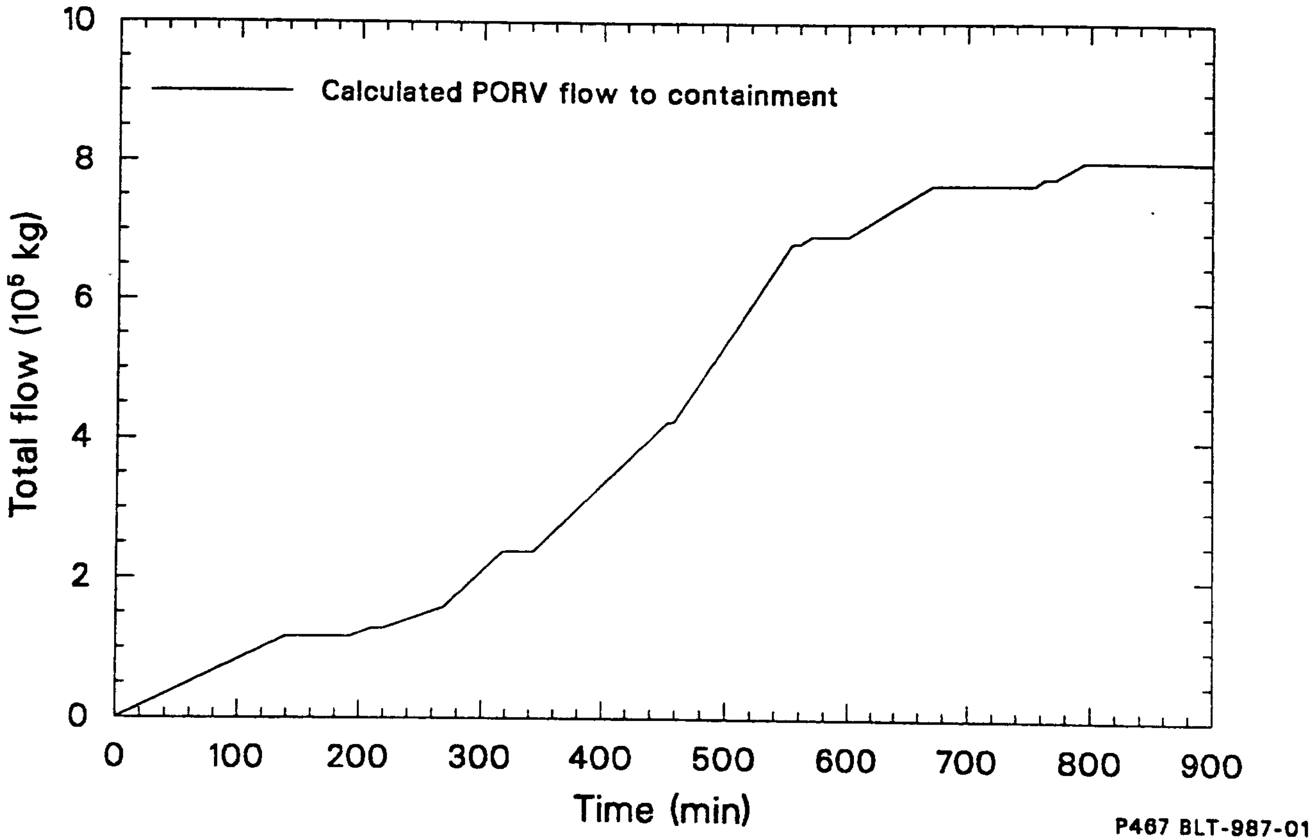

Figure A-10. PORV flow to the RCDT during first 15 hours of the accident.

TABLE A-13. MEASURED ACTIVITY CONCENTRATION AND TOTAL MASS OF THE MAKEUP AND LETDOWN DEMINERALIZER RESINS AND LIQUID

DATA CORRECTED TO: 05/83

VOL,MASS,AREA = 9.3000D+05 g
UNCERTAINTY IN VOL,MASS,AREA = 0.0000D+00 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	1.980D+03 uCi/g	0.000D+00 uCi/g
CS-134	8.720D+02 uCi/g	0.000D+00 uCi/g
CS-137	1.370D+04 uCi/g	0.000D+00 uCi/g

RCS was depressurized. The total volume of coolant that passed through the RCDT during that period was estimated at 7×10^8 ml (Ref. A-11).

From December 1982 until July 1983, the reactor system was again pressurized, resulting in additional leakage to the RCDT of approximately 2×10^8 ml (Ref. A-11).

The total volume of liquid that passed through the RCDT was 1.9×10^9 ml before sampling of the tank. Samples of the RCDT liquid and sediment were collected in December 1983. The sediment samples were taken from the bottom inside surface of the RCDT, directly beneath the rupture disk and vertical section of the rupture line. The volume of RCDT sediment was estimated to be 26 kg as shown in Table A-14. As noted in Ref. A-10, the uncertainty in the estimated RCDT sediment is ±100% due to limited inspection of the tank.

The measured activities of the sediment are summarized in Table 4 of Ref. A-20. These values were used for the inventory calculation of Section 3 and are presented in Table A-14.

2.9 RCS Coolant

As discussed in Section 2 of the report, the major fission product transport medium from the core and RCS was via the RCS coolant flow through the reactor vessel and the PORV. During the first day of the accident, an estimated 8×10^5 kg of RCS coolant was lost through the PORV (see Fig. A-10). During the month following the accident, coolant addition to the RCS was controlled by the decrease in noncondensable gases within the RCS (primarily hydrogen). The RCS makeup was further complicated by loss of letdown flow to the makeup and purification systems and the use of the reactor coolant bleed holdup tanks as a source of RCS coolant makeup. After the first month, continued RCS coolant makeup was required to compensate for RCS and makeup system leakage.

The RCS coolant has been monitored extensively since the accident. The general Cs and Sr activity vs. time are shown in Fig. A-11. The large

TABLE A-14. MEASURED ACTIVITY CONCENTRATION AND TOTAL MASS OF THE RCDT SEDIMENT

DATA CORRECTED TO: 04/84

VOL,MASS,AREA = 2.600D+04 g
UNCERTAINTY IN VOL,MASS,AREA = 5.200D+03 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	1.390D+04 uCi/g	7.000D+02 uCi/g
RU-106	6.100D+01 uCi/g	3.000D+00 uCi/g
SB-125	1.570D+01 uCi/g	8.000D-01 uCi/g
I-129	5.200D-08 uCi/g	4.000D-09 uCi/g
CS-134	5.700D+00 uCi/g	4.000D-01 uCi/g
CS-137	9.700D+01 uCi/g	2.000D+00 uCi/g
CE144	9.600D+01 uCi/g	2.000D+00 uCi/g

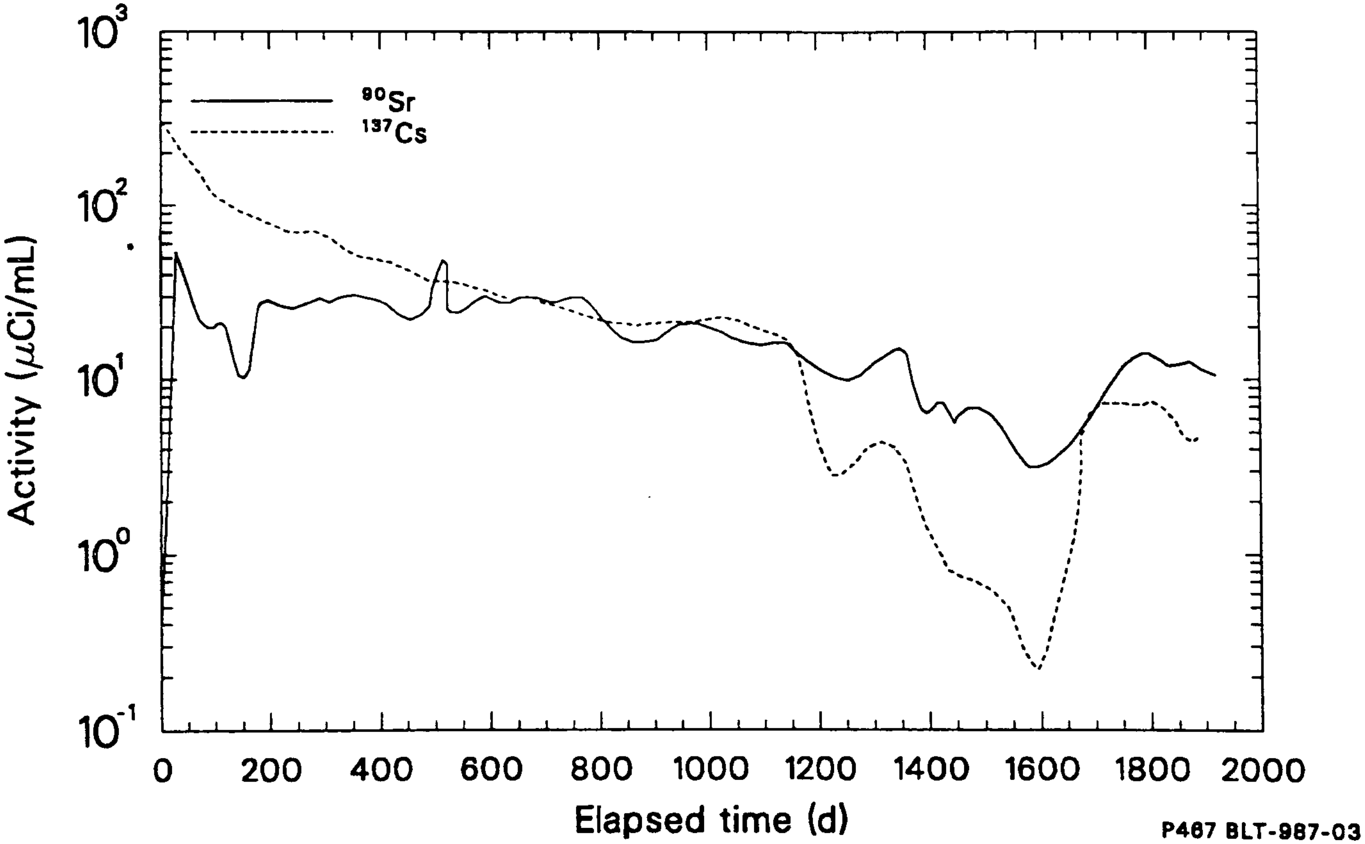

Figure A-11. Reactor coolant system activity vs. time.

decrease in Cs content commencing at 1200 days was due to coolant cleanup activities through the submerged demineralizer system (SDS) (Ref. A-20). The smaller fluctuations resulted from changes in water chemistry and dilution of the RCS coolant due to continued coolant makeup.

Several measurements of the RCS coolant activity have been made; these measurements vary significantly, most likely due to dilution and leakage from the RCS. The data for Sr-90, I-129, Cs-134, and Cs-137 were obtained from Table 10 of Ref. A-22. The data for Sb-125 and Ce-144 were obtained from Table 3-6 of Ref. A-23. These data (used for the inventory calculations of Section 3) are summarized in Table A-15 and represent the measured activities of the RCS coolant samples on August 14, 1980. The total RCS coolant mass associated with the measured activity data were taken from Ref. A-20 and are summarized in Table A-15.

TABLE A-15. MEASURED ACTIVITY CONCENTRATIONS AND TOTAL MASS FOR RCS COOLANT

DATA CORRECTED TO: 08/80

VOL,MASS,AREA = 3.330D+08 ml
UNCERTAINTY IN VOL,MASS,AREA = 0.000D+00 ml

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	2.350D+01 uCi/ml	7.000D-01 uCi/ml
I-129	7.100D-06 uCi/ml	3.000D-07 uCi/ml
CS-134	5.270D+00 uCi/ml	8.000D-02 uCi/ml
CS-137	3.060D+01 uCi/ml	2.000D-01 uCi/ml
SB-125	5.100D-02 uCi/ml	0.000D+00 uCi/ml
CE-144	4.900D-02 uCi/ml	0.000D+00 uCi/ml

3. EX-RCS FISSION PRODUCT REPOSITORIES

The previous two sections summarize the major fission product repositories within the reactor cooling system. The following subsections summarize the data for the following major repositories outside the RCS:

- Reactor building[a] water,
- Reactor building sediment,
- Reactor building lower walls,
- Reactor building upper surfaces,
- Reactor building air space,
- Auxiliary building water, and
- Auxiliary building gaseous release.

3.1 Reactor Building Water

The reactor building basement water is attributed to the following three major sources.

1. Flow of RCS Coolant Through the PORV

   During the first day of the accident, RCS coolant continued to escape to the reactor building basement via the stuck-open PORV block valve until 0630, when it was closed. Additional coolant escaped through the PORV from 0713 to 1700 hours when the block valve was intermittently opened to regulate RCS pressure. An estimated 1×10^6 liters of reactor coolant was released to the basement via this pathway during the first three days.

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a. Also referred to as the containment building.

In addition to the 1×10^6 liters of RCS water released during the first three days of the accident, an average of 29.5 liters/hr flowed through the PORV block valve for more than two years following the accident. This leakage contributed 6.74×10^5 liters of RCS water to the basement water volume. Thus, the total volume of RCS water that escaped to the basement was approximately 1.67×10^6 liters which is about 69% of the total volume of water released to the basement as of September 23, 1981.

2. Reactor Building Spray System

As a result of the hydrogen burn pressure spike that occurred at 1350 hours on the day of the accident, the reactor building spray system was activated and remained on for 5 min 40 sec. During that time, the system discharged an estimated 6.43×10^4 liters of chemically treated water into the reactor building atmosphere. The volume of water discharged by the spray system represents about 3% of the total basement water volume as of September 23, 1981.

3. Flow of River Water

Further increase in the reactor building water level after the accident is attributed to leakage from the reactor building air cooling assembly. An estimated 6.81×10^5 liters of river water leaked into the basement from this source. The river water represents about 28% of the maximum basement water inventory prior to the start of SDS processing in September 1981.

The depth of the water in the R/B basement from May 1979 through December 1983 is shown graphically in Fig. A-12. Before the start of SDS processing on September 23, 1981, the water level had been increasing at a fairly constant rate due to leakage from the RCS and the river water cooling system. However, by the time the gross decontamination experiment commenced six months later, in March 1982, about 2.3×10^6 liters of

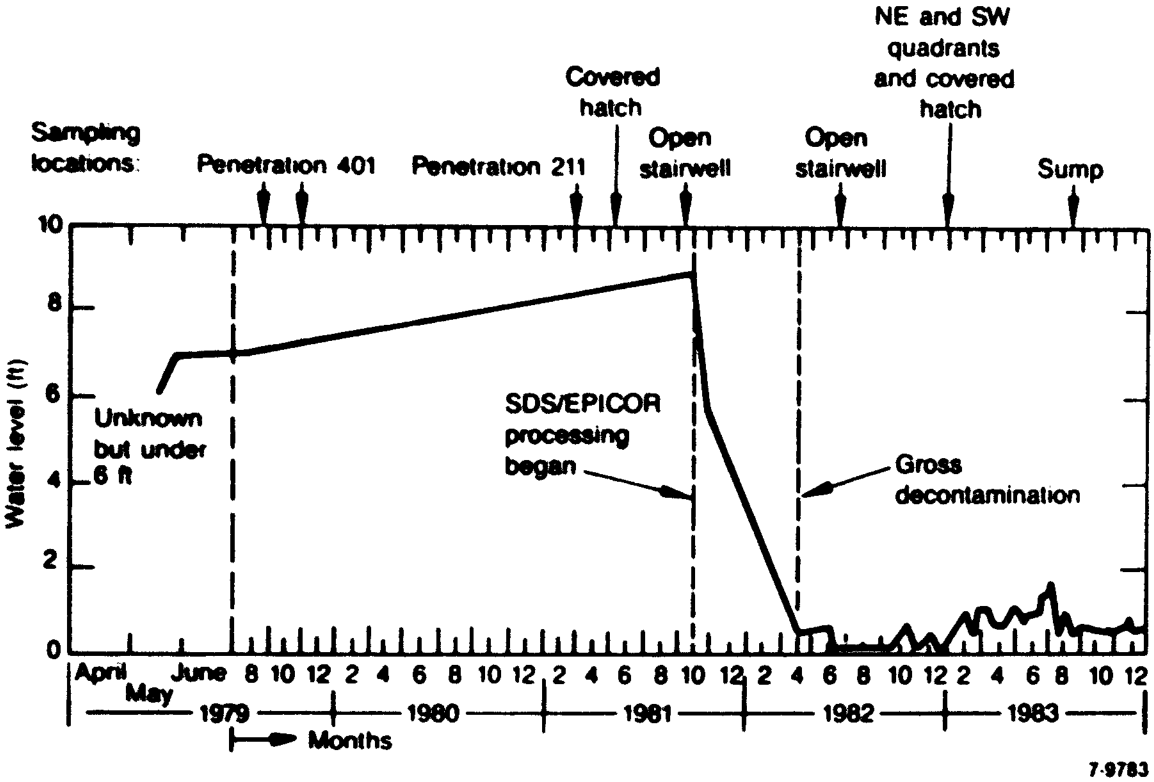


Figure A-12. Reactor building water level vs. time.

contaminated water had been pumped from the basement and processed through the SDS. The gross decontamination experiment and subsequent decontamination operations periodically increased the water depth. By mid-April 1983, an estimated 1.4×10^6 liters of processed water had been used for decontamination purposes and had been returned to the basement.

Table A-16 summarizes the liquid samples that have been collected from the R/B basement since August 1979. Fig. A-13 shows the locations where the liquid samples were collected.

Radiochemical analysis results for the samples that were collected from the reactor building basement from August 1979 through January 1983 are summarized in Table 5 of Ref. A-20. Gamma spectrometer measurements and I-129 and Sr-90 analyses results are presented in Tables 2 and 3 of Ref. A-24. The average radionuclide concentrations as of May 14, 1981, (before SDS processing began on September 23, 1981), were taken as representative for the inventory calculation and are summarized in Table A-17. The associated liquid volume is also shown in Table A-17 and was taken from Table 5 of Ref. A-20.

3.2 Reactor Building Sediment

Reactor building sediment samples were obtained at the same times as the reactor building basement liquid samples.

The radiochemical analyses results for the samples collected from August 1979 through January 1983 are summarized in Table 6 of Ref. A-20. The average radionuclide concentrations as of May 14, 1981 (before SDS processing began) were taken as representative for the inventory calculation and are summarized in Table A-18. The samples were collected from three depths in the water and from the basement floor directly beneath the covered hatch on the 305-ft elevation on May 14, 1981, using the multilevel sampler.

Collectively, the visual inspections of the basement floor indicate that the sediment thickness ranges from 0 to 1.3 cm. For the purpose of

TABLE A-16. SUMMARY OF REACTOR BUILDING BASEMENT LIQUID SAMPLES

Sample Collection Date	Location	Number of Samples Obtained	Quantity (mL)	Physical Character	Sampling Technique Used	Laboratory[a]
08/28/79	Penetration 401, 292-ft elevation	3	30 ea.	Water, slurry	Liquid suction through tygon tubing	ORNL
11/15/79	Penetration 401	1	1050	Slurry	Liquid suction through tygon tubing	ORNL
03/19/81	Penetration 211	3	1000 ea.	Water	Liquid suction through tygon tubing	ORNL
05/14/81	Covered equipment hatch, 305-ft elevation	8	~100 ea.	Water, slurry	Vacuum actuated, plunger-operated sampler	INEL
09/24/81	Open stairwell, 305-ft elevation	1	~100	Slurry	Vacuum actuated, plunger-operated sampler	INEL
06/23/82	Bottom of open stairwell	1	~45	Slurry	Manual scoop	ORNL, WHEDL
01/11/83	Covered equipment hatch, 305-ft elevation	1	45	Slurry	Remotely operated, solenoid-actuated sampler	PNL/TMI
01/11/83	Northeast quadrant, penetration 238	1	55	Slurry	Remotely operated, solenoid-actuated sampler	PNL/TMI
01/11/83	Southwest quadrant, penetration 225	1	55	Slurry	Remotely operated, solenoid-actuated sampler	PNL/TMI
08/22/83	Sump pump discharge line	2	300 ea.	Slurry	On-line sampling bomb	INEL, WHEDL
12/05, 12/83	Reactor coolant drain tank	2	~120 ea.	Water, slurry	Vacuum actuated, plunger-operated sampler	INEL, WHEDL

a. ORNL: Oak Ridge National Laboratory; INEL: Idaho National Engineering Laboratory; WHEDL: Westinghouse Hanford Engineering Development Laboratory; PNL/TMI: Pacific Northwest Laboratory Mobile Facility at TMI.

TABLE A-17. MEASURED ACTIVITY CONCENTRATION AND SAMPLE VOLUME/MASS FROM THE REACTOR BUILDING LIQUID

DATA CORRECTED TO: 05/81

VOL,MASS,AREA = 2.390D+09 ml
UNCERTAINTY IN VOL,MASS,AREA = 0.000D+00 ml

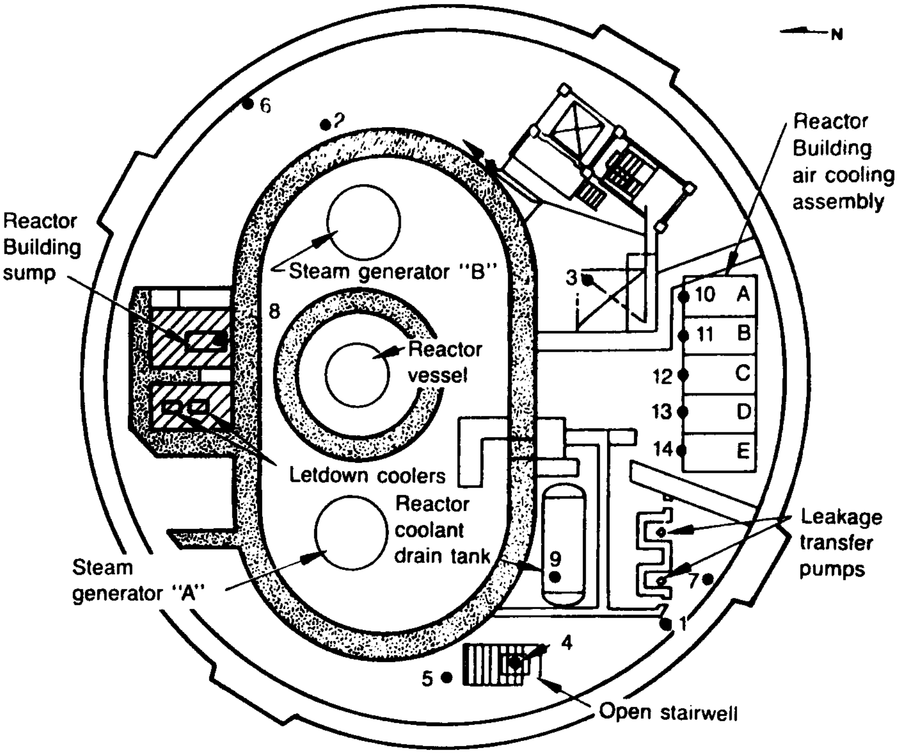
ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	5.200D+00 uCi/ml	3.000D-01 uCi/ml
SB-125	3.000D-02 uCi/ml	3.000D-03 uCi/ml
I-129	4.300D-06 uCi/ml	3.000D-07 uCi/ml
CS-134	1.920D+01 uCi/ml	1.000D-01 uCi/ml
CS-137	1.430D+02 uCi/ml	1.000D+00 uCi/ml

TABLE A-18. MEASURED ACTIVITY CONCENTRATION AND ASSOCIATED MASS OF THE REACTOR BUILDING (CONTAINMENT) SEDIMENT

DATA CORRECTED TO: 05/81

VOL,MASS,AREA = 3.7600D+05 g
UNCERTAINTY IN VOL,MASS,AREA = 0.0000D+00 g

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	8.0000D+02 uCi/g	2.0000D+02 uCi/g
RU-106	1.0400D+02 uCi/g	7.0000D+00 uCi/g
SB-125	4.8700D+02 uCi/g	9.0000D+00 uCi/g
I-129	1.1000D-01 uCi/g	1.0000D-02 uCi/g
CS-134	1.0700D+02 uCi/g	1.0000D+00 uCi/g
CS-137	8.0000D+02 uCi/g	3.0000D+00 uCi/g
CE-144	6.6000D+01 uCi/g	3.0000D+00 uCi/g



Sampling locations	
1	Penetration 401 292-ft elevation
2	Northeast quadrant near penetration 211
3	Covered equipment hatch 305-ft elevation
4	Open stairwell
5	Bottom of open stairwell
6	Northeast quadrant penetration 238
7	Southwest quadrant penetration 225
8	Sump pump discharge line
9	Reactor coolant drain tank
10-14	Reactor Building air cooling assembly 305-ft elevation

7-9787

Figure A-13. Reactor building basement water sample locations.

these calculations, it is assumed that the average thickness of the sediment layer covering the basement floor is 0.035 cm and that the solid density of the sediment layers is 63.5 mg/cm^3. The surface area of the basement floor is 9.11×10^6 cm^2. Therefore the associated mass of sediment on the basement floor is 367 kg. This mass is shown in Table A-18 and is taken from Table 6 of Ref. A-20.

3.3 Reactor Building Lower Walls (Below High-water Level)

Three concrete samples were removed from the reactor building basement walls at the location shown in Fig. A-14. Locations 'A1' and 'sub-2' were below the water level from approximately January through October 1981. However, the B2 sample was exposed to the accident water from the beginning of the accident until water processing was nearly completed (i.e., January 1982). The samples represent the following three categories of concrete surfaces that were submerged in water from the accident:

1. Unpainted, 3000 psi concrete (Location A1)

2. Painted, 5000 psi concrete (Location B2)

3. Unpainted, concrete block (Location sub-2).

Table 3 of Ref. A-25 lists the total quantity of fission products present in each concrete sample and the total percentages leached into solution during the four-month study. The surface concentration of the concrete wall used for the inventory calculation was obtained by summing the retained and leached components of the activity and dividing by the surface area of the samples. These data are shown in Table A-19.

A list of reactor building surface areas is summarized in Table 55 of Ref. A-26. The total surface area of immersed basement walls is calculated to 2.90×10^7 cm^2 based on an estimated maximum water depth in the basement of 2.6 m during and after the accident. This surface area is noted in Table A-19.

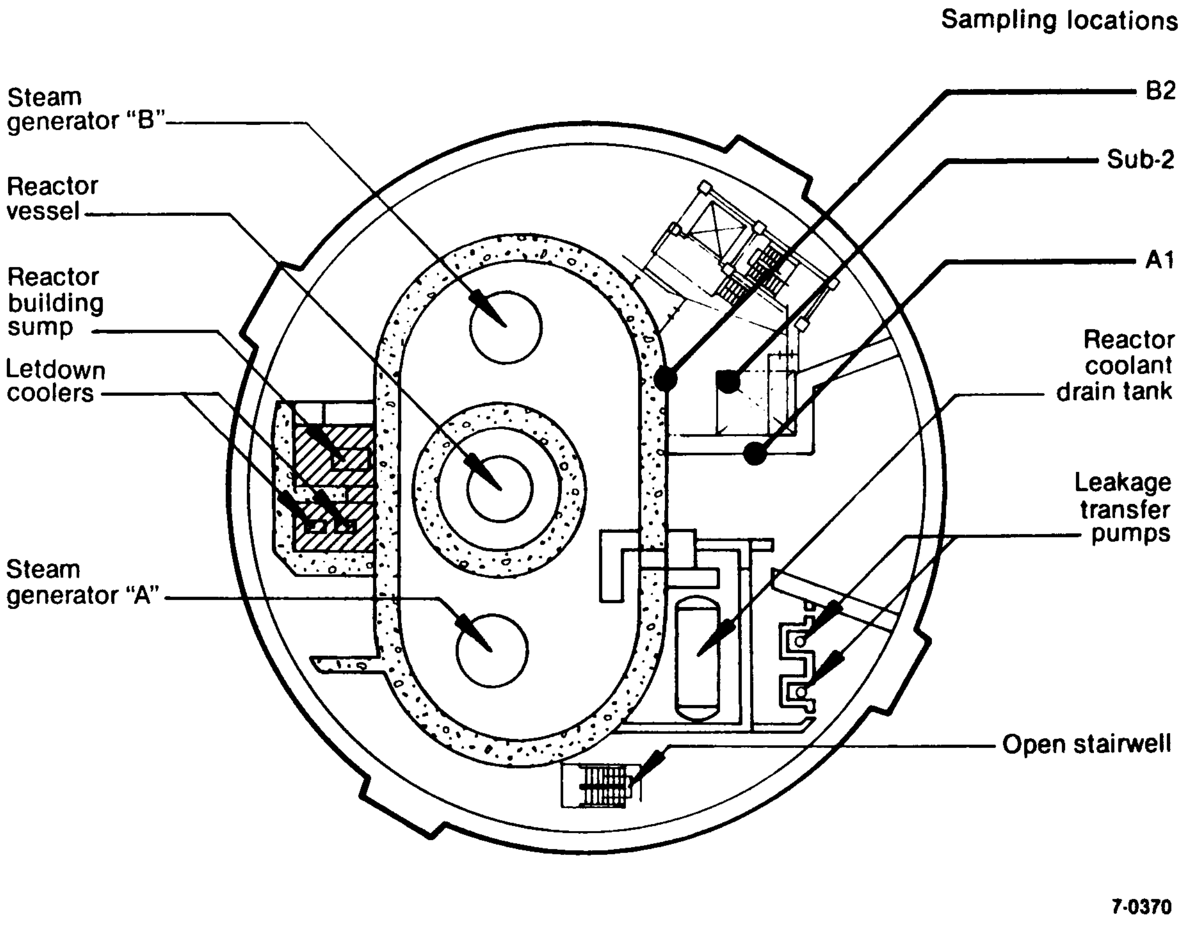


Figure A-14. Location of reactor basement wall samples.

TABLE A-19. MEASURED ACTIVITY CONCENTRATION AND ASSOCIATED SURFACE AREA OF THE REACTOR BUILDING BASEMENT WALLS (BELOW WATER LEVEL)

DATA CORRECTED TO: 03/86

VOL,MASS,AREA = 2.9000+07 cm2
UNCERTAINTY IN VOL,MASS,AREA = 0.0000+00 cm2

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	6.0400+01 uCi/cm2	0.0000+00 uCi/cm2
CS-137	2.9500+03 uCi/cm2	0.0000+00 uCi/cm2

3.4 Reactor Building Upper Surfaces (Above High-water Level)

In-situ gamma scans of the reactor building air cooling assembly (11C, 11D, and 11E cooling coils and drip pans) were completed in October 1981 (see Fig. A-15). In addition, 85 samples were obtained from the reactor building structural surfaces in December 1981 (before the gross-decontamination experiment) and an additional 95 surface samples were obtained from the same surfaces in late March 1982, following the completion of the gross-decontamination experiment.

In April 1983 a total of five panels were removed from the reactor building air coolers (one from each reactor building air cooler). The panels were shipped to the INEL for laboratory examination to characterize surface depositions.

Results of the in-situ gamma spectral measurements of the reactor building cooling assembly surfaces are presented in Table 27 of Ref. A-20. Results of the surface activity measurements performed at the INEL are presented in Tables 24 and 25 of the same reference. Because the gamma scan measurements of the drip pans and cooling coils did not yield results for Sr-90 and I-129, the air cooler access panel analyses, rather than the measurements of the pans and coils, were used to represent the containment surface deposition for the inventory calculations. These data are shown in Table A-20. Table 26 of Ref. A-20 shows the comparison of the air cooler access panel data with data from surface samples obtained from the 305 ft elevation of the reactor building. The comparison suggests that deposition was similar for all radionuclides at both locations.

Table 2 of Ref. A-20 summarizes the surface areas of components associated with the reactor building air coolers. Table 55 of Ref. A-26 documents the total internal reactor building surface area. The reactor building surface area above the water level, used for the inventory calculation, is shown in Table A-20.

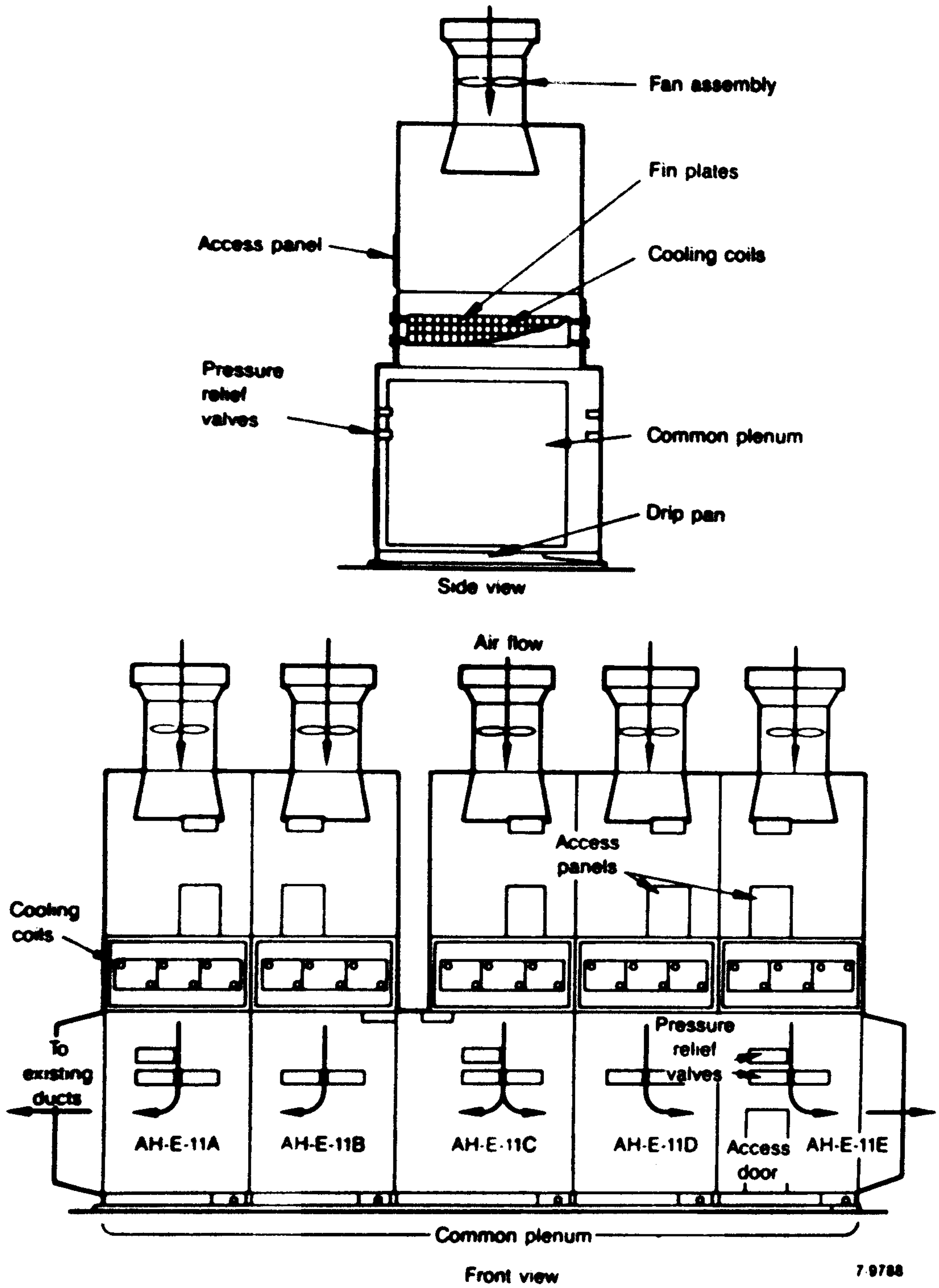


Figure A-15. Location of the reactor building air cooling assemblies.

TABLE A-20. MEASURED ACTIVITY CONCENTRATION AND ASSOCIATED SURFACE AREA OF THE REACTOR BUILDING UPPER WALLS

DATA CORRECTED TO: 03/84

VOL,MASS,AREA = 1.910D+08 cm2
UNCERTAINTY IN VOL,MASS,AREA = 0.000D+00 cm2

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	1.800D-02 uCi/cm2	1.700D-02 uCi/cm2
I-129	5.600D-06 uCi/cm2	2.200D-06 uCi/cm2
CS-134	4.300D-02 uCi/cm2	4.500D-02 uCi/cm2
CS-137	7.600D-01 uCi/cm2	8.000D-01 uCi/cm2

3.5 Reactor Building Air Space

During the period from April 29, 1980, to May 2, 1980, the atmosphere of the TMI-2 reactor building was sampled and subsequently analyzed to provide characterization before the containment purge.

Analyses for the determination of I-129, C-14, H-3, Kr-85, and radionuclide activity of suspended particulates and for molecular analysis of the atmosphere gaseous components were performed by two different analytical laboratories at INEL.

The radionuclide concentrations determined during the sampling program and estimated total containment free volume associated with the gas sample data are presented in Table 14 of Ref. A-27 and are summarized in Table A-21.

3.6 Auxiliary Building Liquid

The radioactive coolant in the auxiliary building consisted of:

- inventory existing before the accident,
- contaminated water transferred from the reactor containment building sump to the auxiliary building during the early phases of the accident,
- letdown from the reactor coolant system, and
- normal leakage from system components.

Approximately 280,000 gallons (1050 m^3) of intermediate-level waste exists in the auxiliary building tanks. The radioactive water volume in each auxiliary building tank is indicated in Table II-8 of Ref. A-28 and the radioactive concentration on June 15, 1979 is tabulated in Table II-9 of the same reference.

TABLE A-21. MEASURED VOLUMETRIC ACTIVITY AND ASSOCIATED FREE VOLUME OF THE REACTOR BUILDING (CONTAINMENT)

DATA CORRECTED TO: 04/80

VOL,MASS,AREA = 5.580D+10 cm3
UNCERTAINTY IN VOL,MASS,AREA = 0.000D+00 cm3

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	1.600D-10 uCi/cm3	3.000D-11 uCi/cm3
RU-106	9.000D-11 uCi/cm3	0.000D+00 uCi/cm3
SB-125	2.000D-10 uCi/cm3	0.000D+00 uCi/cm3
I-129	5.700D-11 uCi/cm3	4.000D-12 uCi/cm3
CS-134	1.100D-10 uCi/cm3	1.000D-11 uCi/cm3
CS-137	7.200D-10 uCi/cm3	8.000D-11 uCi/cm3
CE-144	8.000D-11 uCi/cm3	0.000D+00 uCi/cm3
EU-154	2.000D-11 uCi/cm3	0.000D+00 uCi/cm3
EU-155	3.000D-11 uCi/cm3	0.000D+00 uCi/cm3
KR-85	8.800D-01 uCi/cm3	4.000D-02 uCi/cm3

The reactor coolant bleed tank has the in-flow path from the liquid relief valve of the makeup tank. Two days after the accident, the liquid relief valve on the makeup tank was opened, allowing the highly radioactive contents in the tank to flow into the reactor coolant bleed tanks.

The water volume in the reactor coolant bleed tanks makes up approximately 85% of total radwaste volume in the auxiliary building tanks, and the fission product concentration in these tanks is several times higher than those in the other tanks. So the fission product inventory in the reactor coolant bleed tanks can be used as the representative repository for the auxiliary building liquid.

The three liquid reactor coolant bleed tank samples, identified as RCBT-A, RCBT-B, and RCBT-C were collected from Tanks A, B, and C on December 20, 1979, January 28, 1980 and February 4, 1980, respectively. These bleed tank samples were taken prior to processing through EPICOR-II. The results are listed in Table 10 of Ref A-22. Table A-22 summarizes the concentration and associated liquid volume for the reactor coolant bleed tanks used for the inventory calculations of Section 3. The concentration data were obtained by averaging the data in Table 10 of Ref. A-23. The liquid volume was obtained from page 4-3 of Ref. A-29.

3.7 Auxiliary Building Gas Release

The radioactive materials released to the environment as a result of the TMI-2 accident were those that escaped from the damaged fuel and were transported in the coolant via the letdown line into the auxiliary building and then into the environment. The noble gases and radioiodines, because of their volatile nature and extensive release from the fuel, were the primary radionuclies released from the auxiliary building to the environment.

The principal release of radioactive noble gases occurred on the first day of the accident. The total quantity of released radioactive materials is estimated as 2.5 million Ci. Table II-1 of Ref. A-28 summarizes the estimated quantity released of Kr-88, Xe-133, X3-133m, Xe-135, Xe-135m, and I-131.

TABLE A-22. MEASURED ACTIVITY AND ASSOCIATED VOLUME OF REACTOR COOLANT BLEED TANKS

DATA CORRECTED TO: 05/81

VOL,MASS,AREA = 9.460D+08 ml
UNCERTAINTY IN VOL,MASS,AREA = 7.000D+07 ml

ISOTOPE	CONCENTRATION	UNCERT. IN CONCENTRATION
SR-90	4.300D-01 uCi/ml	7.000D-02 uCi/ml
I-129	1.200D-05 uCi/ml	1.000D-06 uCi/ml
CS-134	8.850D+00 uCi/ml	9.100D-01 uCi/ml
CS-137	4.370D+01 uCi/ml	4.000D-01 uCi/ml

Kr-85 and I-129 are not included in Table II-1 of Ref. A-27 because the released concentrations were below the detectability limits. Based on the data in Table II-1 of Ref. A-27, the maximum release fraction for the noble gases is 0.01; for Iodine-131 the release fraction is $\sim 2 \times 10^{-7}$. Assuming these release fractions are valid for Kr-85 and I-129, the total release of Kr-85 and I-129 is 9.69×10^{2} Ci and 4.3×10^{-8} Ci, respectively. The estimated environmental gas release from auxiliary building used for inventory calculations is presented in Table A-23.

TABLE A-23. ESTIMATED ENVIRONMENTAL RELEASE OF NOBLE GASES AND IODINE (from Ref. A-27)

Isotope	Activity Release to the Environment (Ci)
Kr-85	9.69×10^{2}
Kr-88	3.75×10^{5}
Xe-133	1.58×10^{6}
Xe-133m	2.25×10^{5}
Xe-135	3.0×10^{5}
Xe-135m	2.5×10^{4}
I-129	4.3×10^{-8}
I-131	15

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APPENDIX B

INTACT ROD FISSION PRODUCT INVENTORY

FIGURES

TABLES

APPENDIX B
INTACT ROD FISSION PRODUCT INVENTORY

As noted in Section 3 and Appendix A, the degraded core regions have been identified from video inspections. This Appendix presents the data and calculational results for estimating the fission product inventory contained within the intact rod region (end-state).

It is assumed that the relative core fission product distribution is identical to the core burnup distribution. The core burnup data (determined from the incore neutron monitors) defines a local burnup value for each of seven equal axial regions for each core fuel assembly (Ref. B-1). The burnup data is summarized in Table B-1. Figure B-1 is a core cross-section showing the fuel assembly identifications.

The burnup and end-state core configuration data were used to calculate the fraction of the total core burnup within the end-state region of intact rods using the following equations:

$$\text{Fractional Burnup in Intact Rods} = \sum_{i=1}^{177}\sum_{j=1}^{7} (BU)_{ij}F_{ij}$$

where

i = fuel assembly number,

j = axial burnup region,

BU_{ij} = burnup in region i, j, and

F_{ij} = fraction of rods intact in region i, j.

The fraction of each fuel assembly axial region containing intact rods, F_{ij}, was estimated from Figs. B-2 through B-14 showing the upper and lower bounds of the intact rod regions. These configuration data are based on the core bore inspection data evaluation summarized in Ref. B-2. As noted in these figures, two cases are considered: the first represents an upper bound region of intact rods, the second represents a lower bound region of intact rods. The uncertainty in the configurations is based on uncertainty in interpreting the core bore inspection data. The intact rod fractions for each fuel assembly axial region[a] are tabulated in Tables B-2 and B-3 for the upper and lower bound cases, respectively.

Table B-4 summarizes the results of calculations estimating the fractional core burnup within the intact rod region. The results indicate that the fractional core burnup within the intact rod zone (fraction fission product inventory) is approximately 35% and 30% for the upper and lower bound cases, respectively. Based on these results, the fractional core fission product inventory of the intact rods is assumed to be 32.5 ± 2.5%.

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a. All axial regions are relative to the top of the fuel rods, i.e., Region #1 is at the top of the rod, Region #7 is at the bottom of the rod.

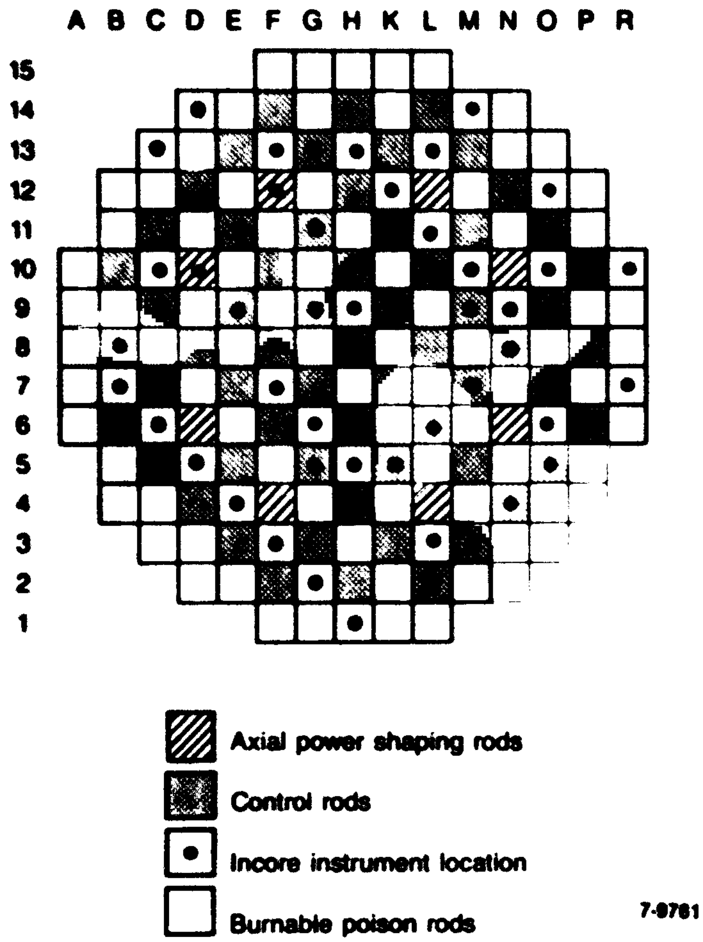


Figure B-1. TMI-2 fuel assembly identification.

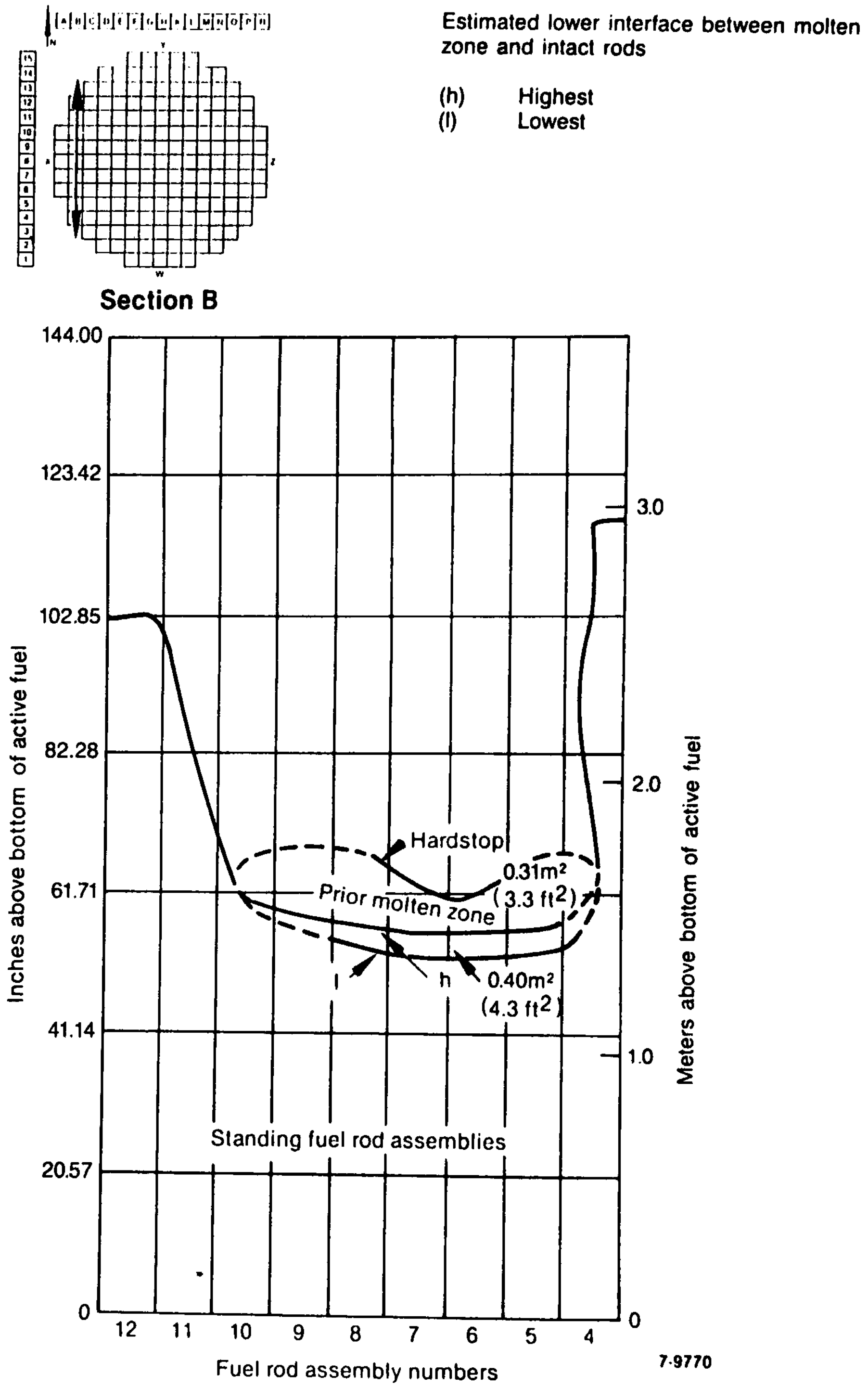


Figure B-2. End-state bounds for intact rods (B-column of fuel assemblies).

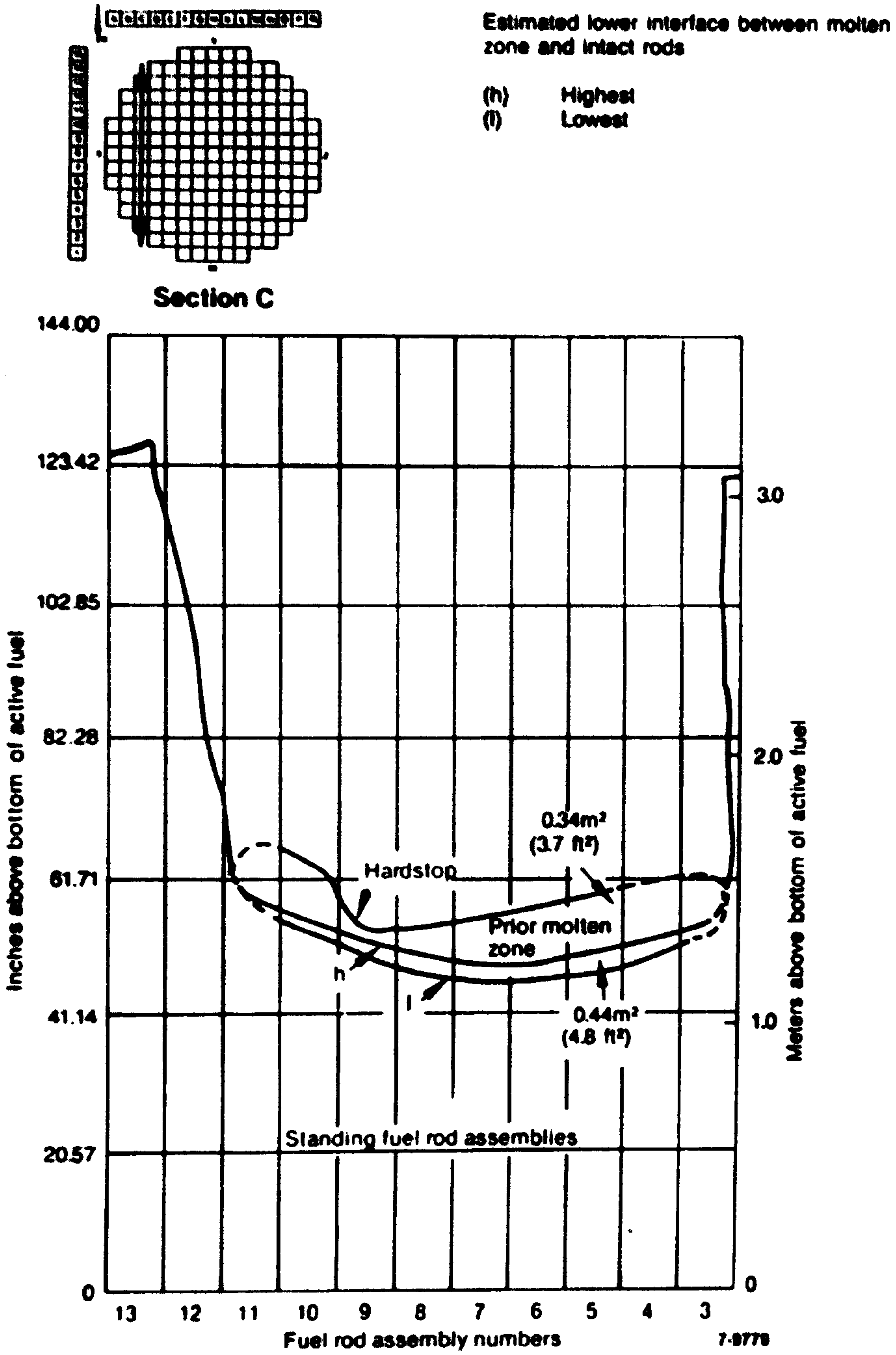


Figure B-3. End-state bounds for intact rods (C-column of fuel assemblies).

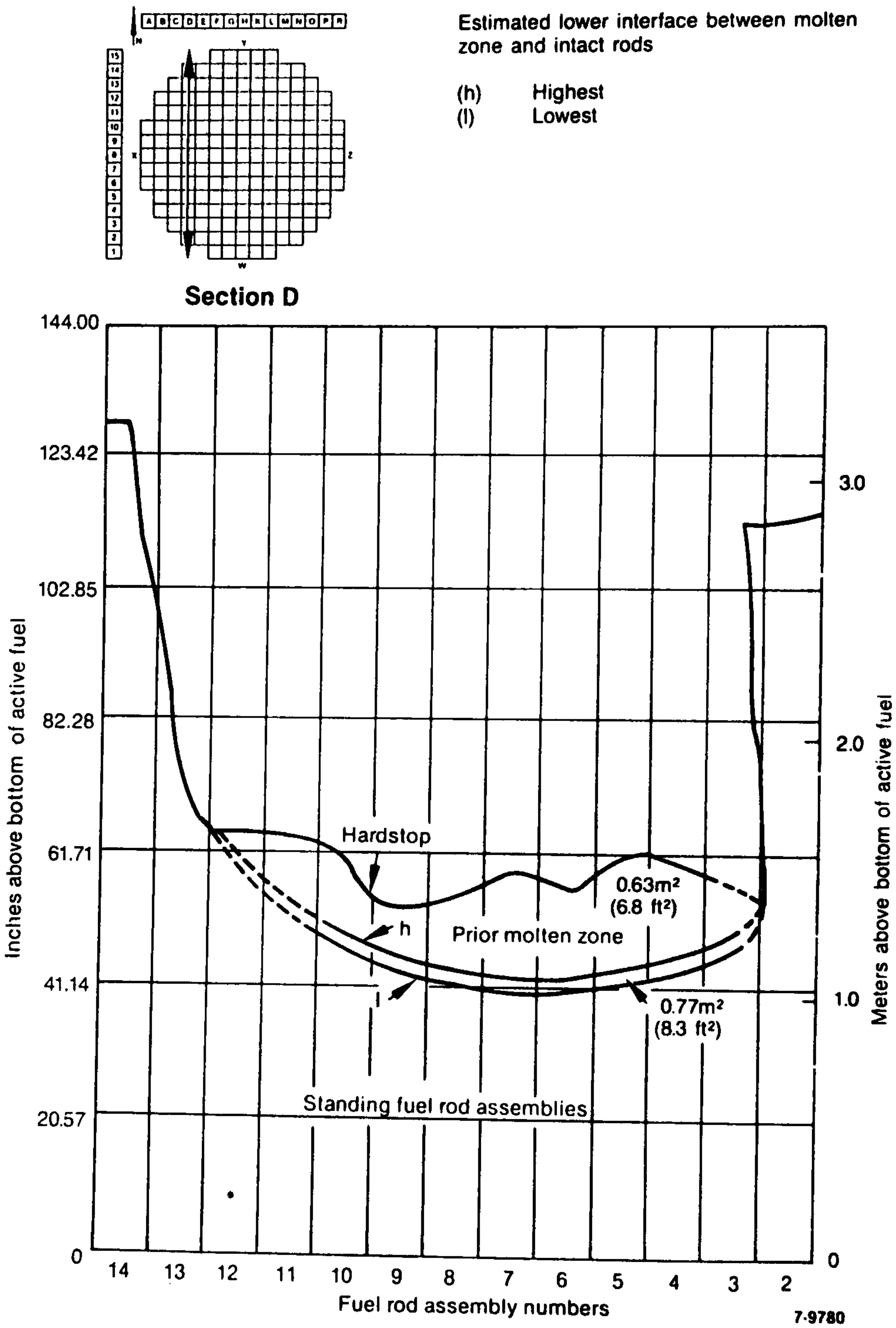


Figure B-4. End-state bounds for intact rods (D-column of fuel assemblies).

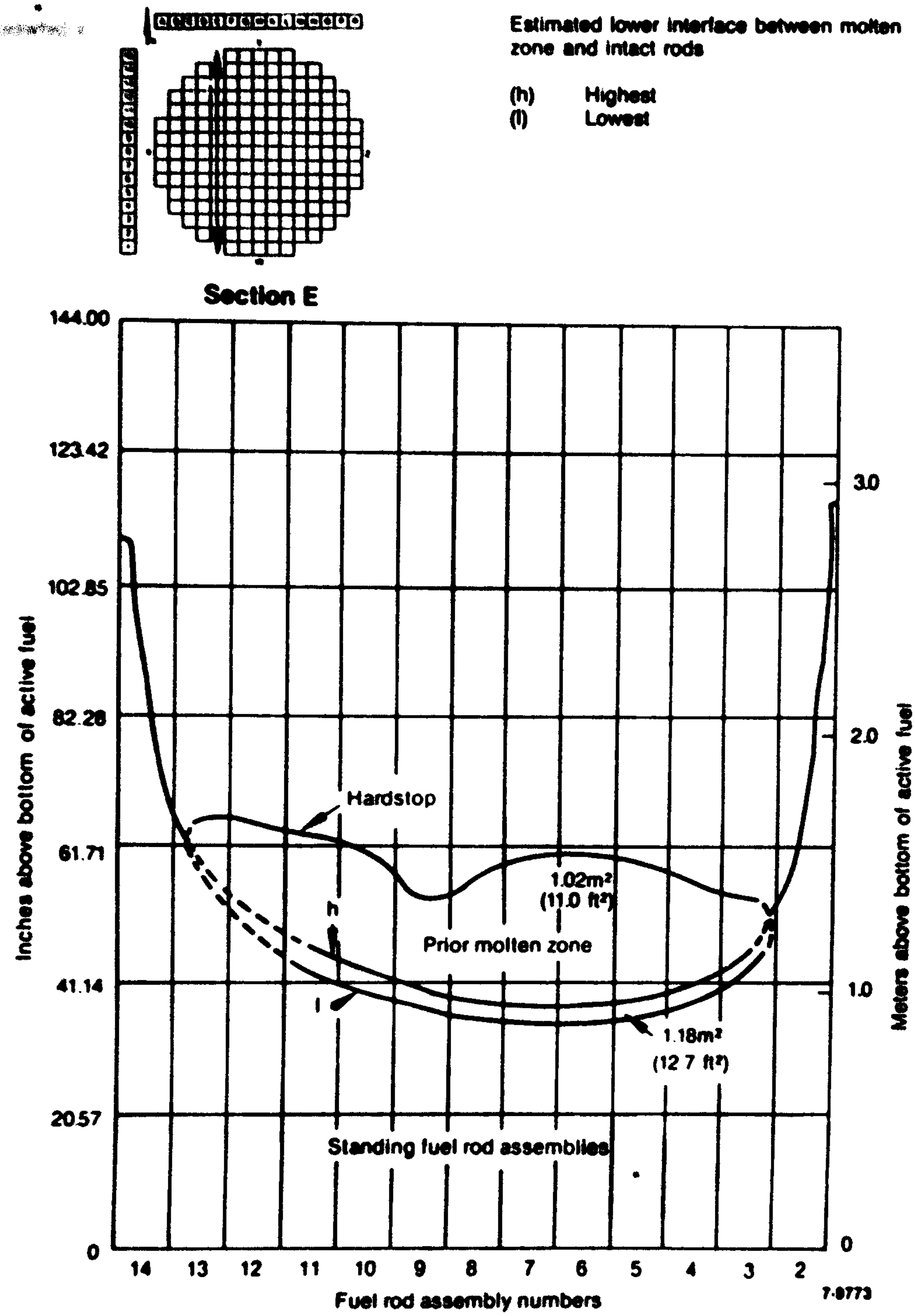


Figure 8-5. End-state bounds for intact rods (E-column of fuel assemblies).

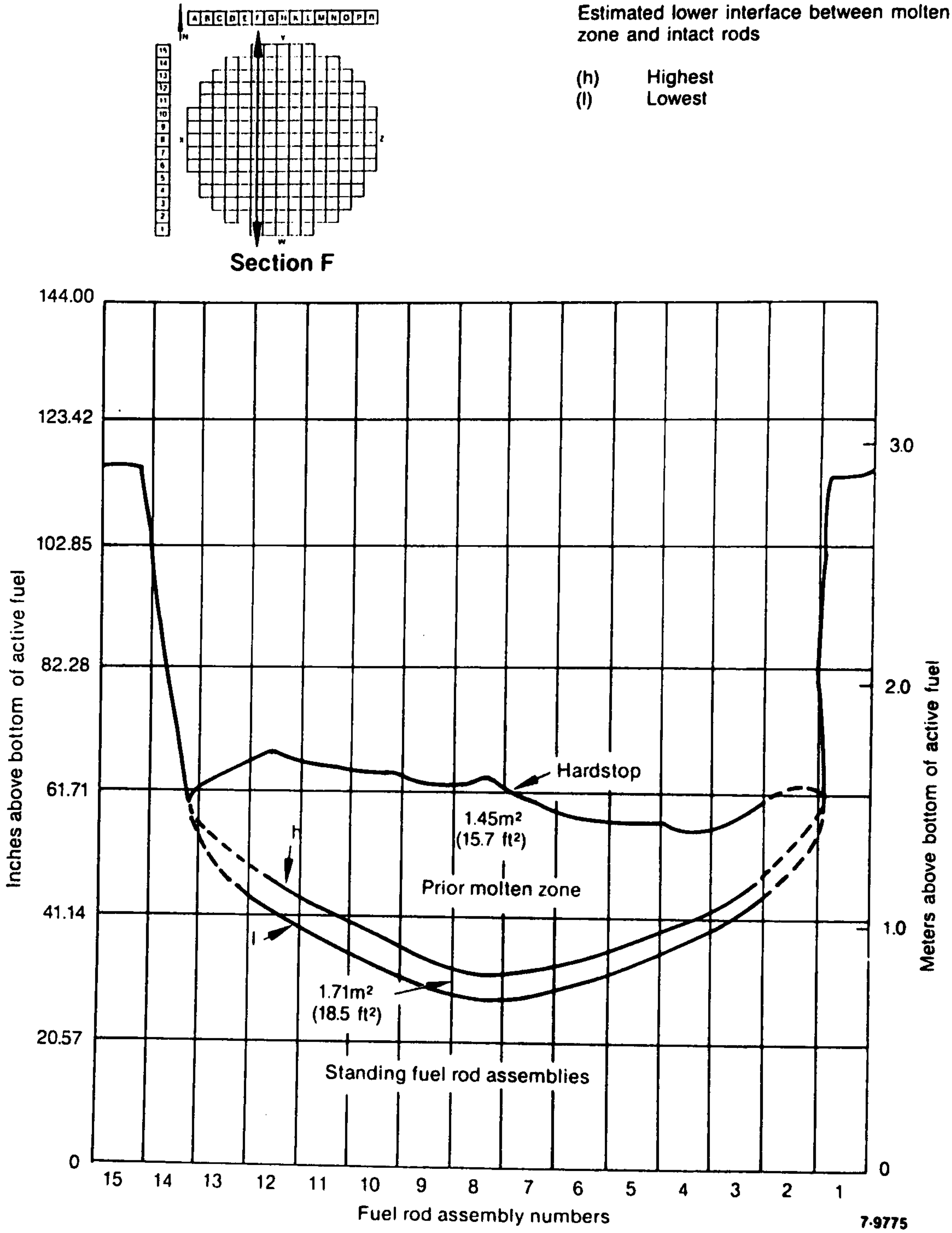


Figure B-6. End-state bounds for intact rods (F-column of fuel assemblies).

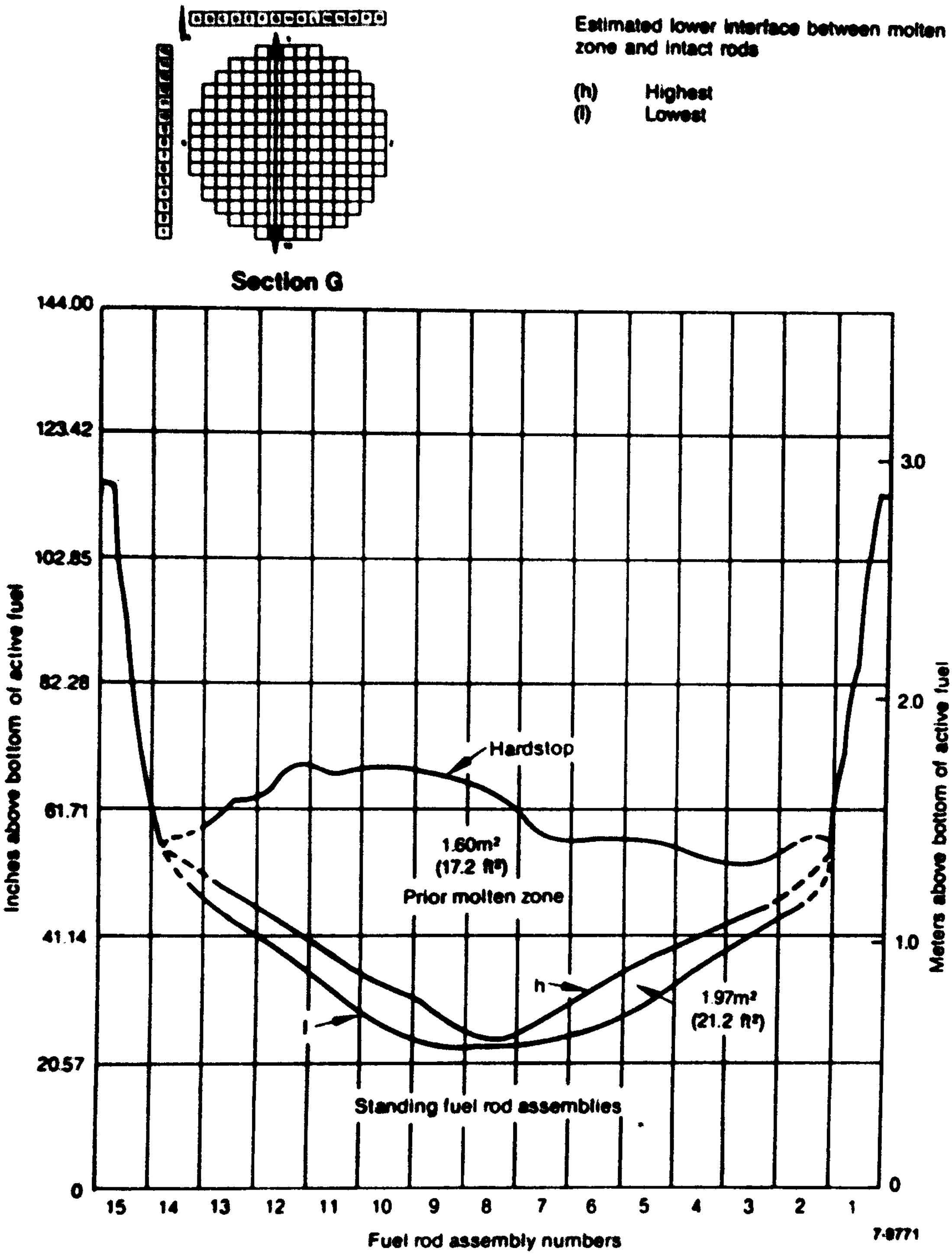


Figure B-7. End-state bounds for intact rods (G-column of fuel assemblies).

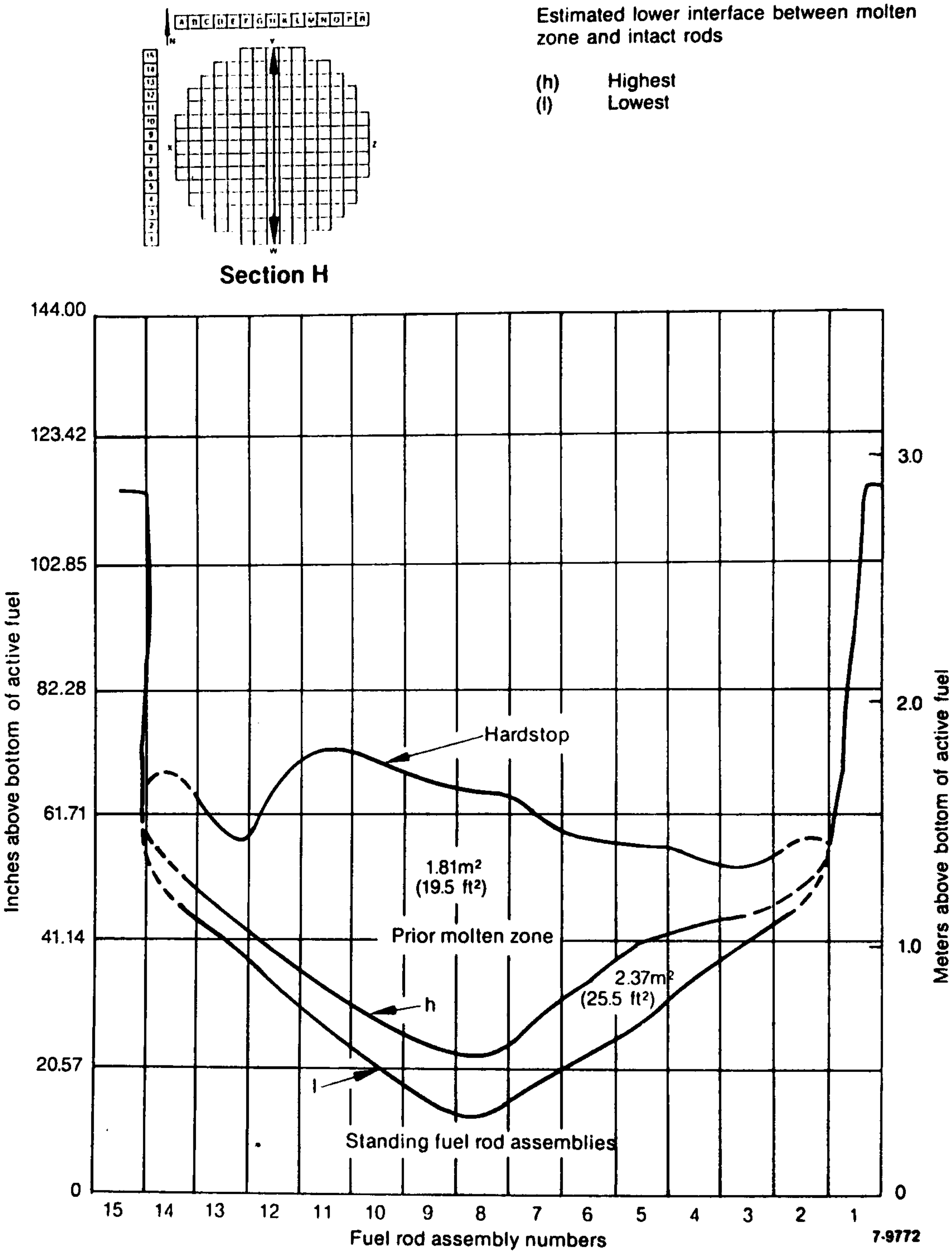


Figure B-8. End-state bounds for intact rods (H-column of fuel assemblies).

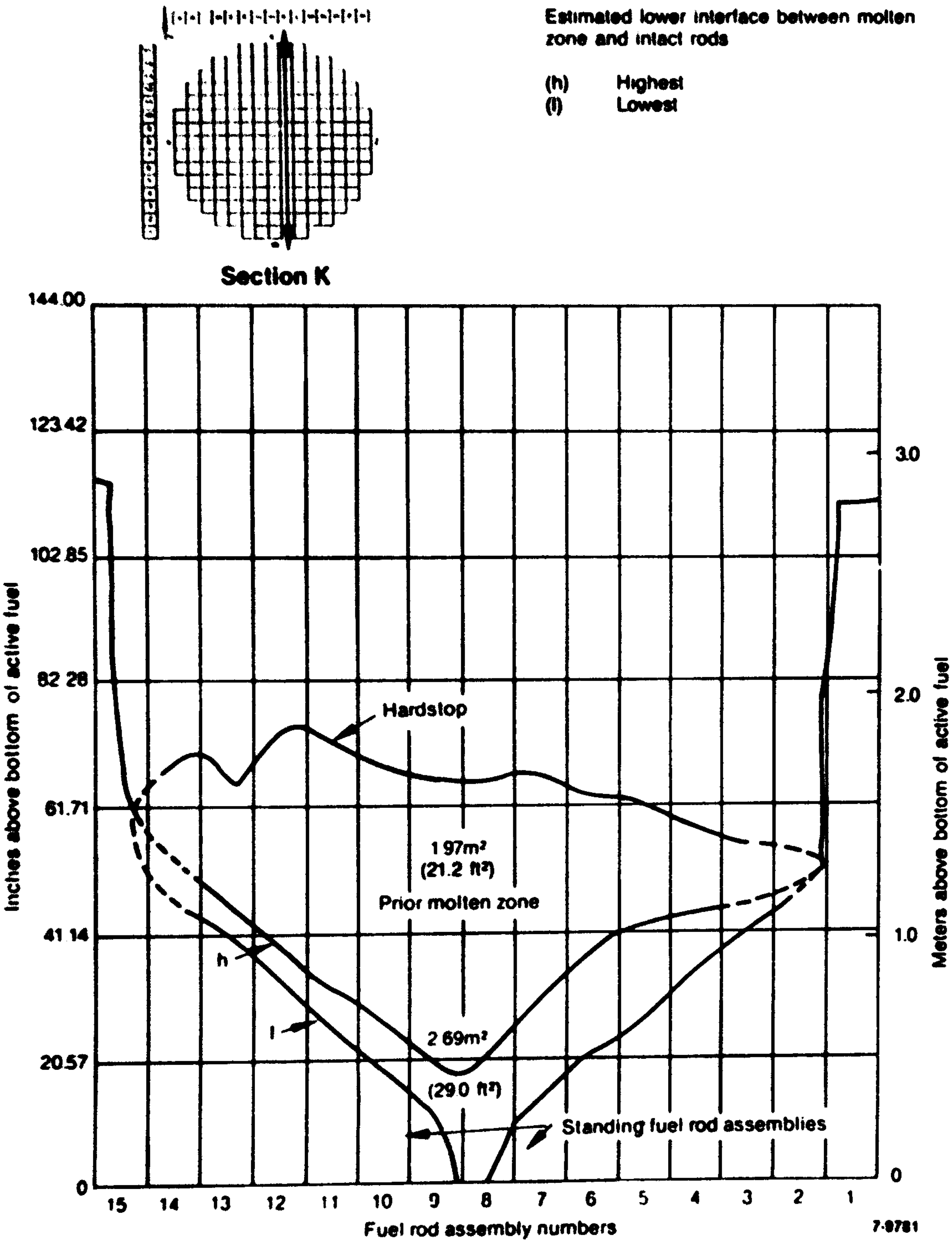


Figure B-9. End-state bounds for intact rods (K-column of fuel assemblies).

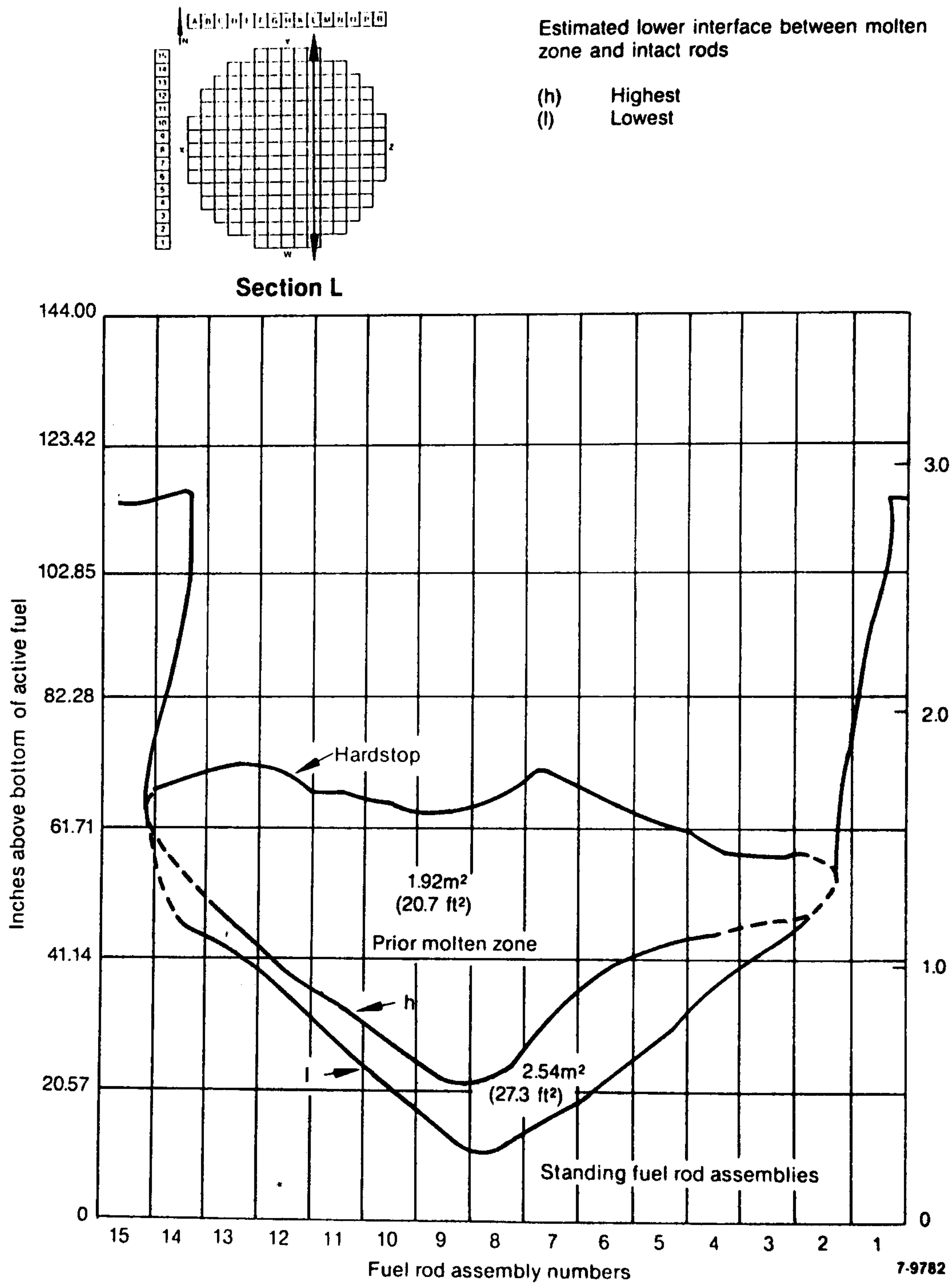


Figure B-10. End-state bounds for intact rods (L-column of fuel assemblies).

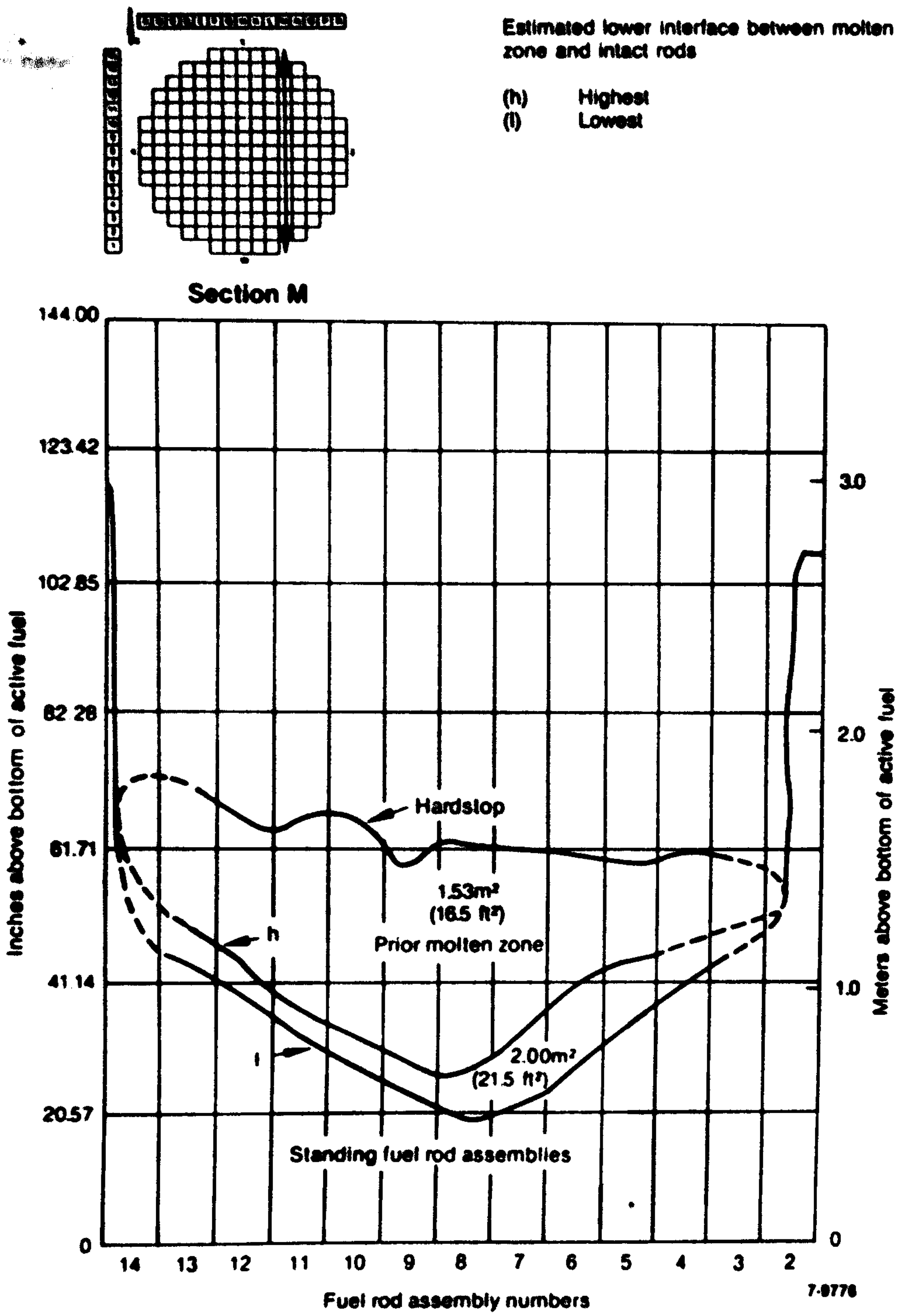


Figure B-11. End-state bounds for intact rods (M-column of fuel assemblies).

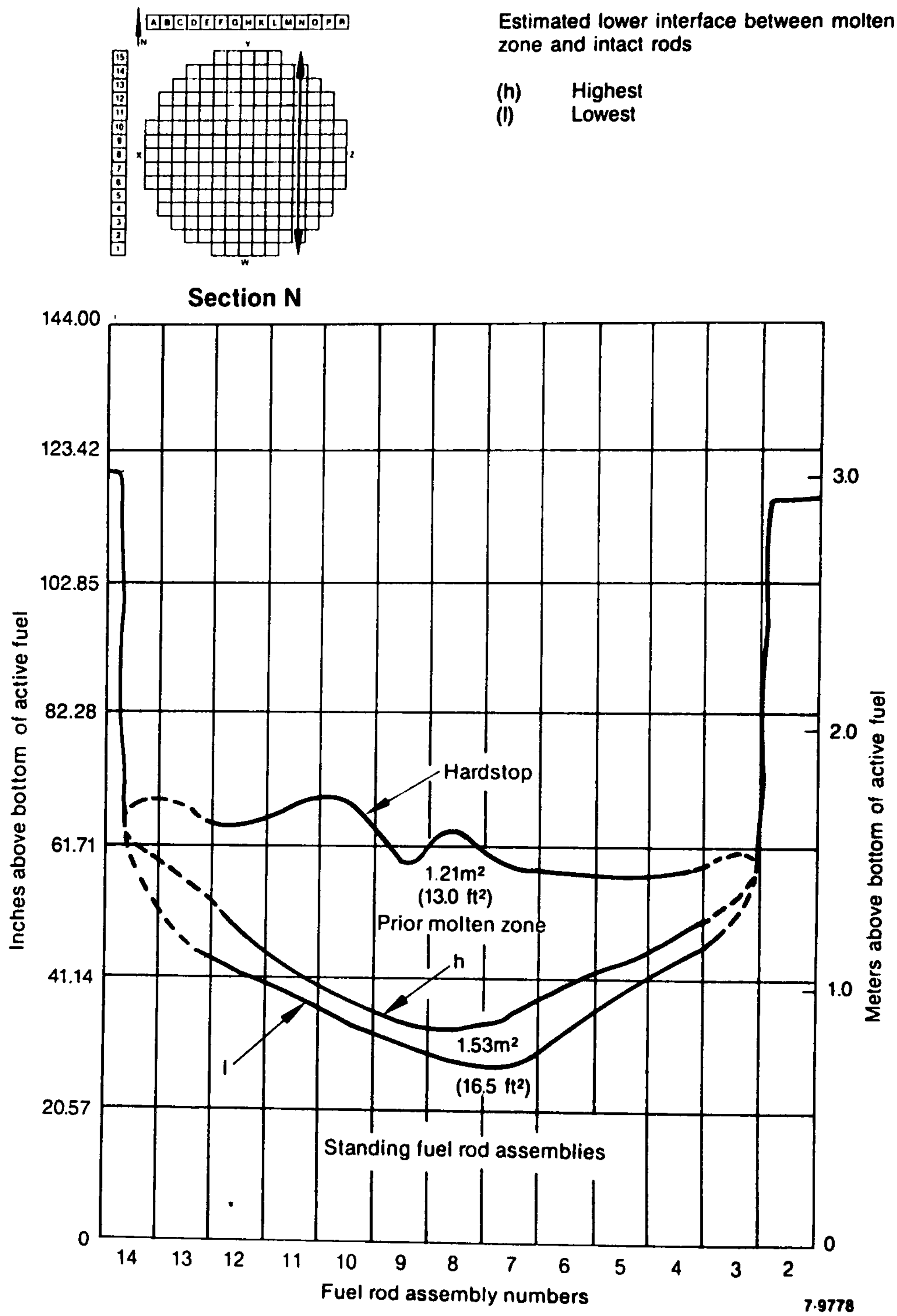


Figure B-12. End-state bounds for intact rods (N-column of fuel assemblies).

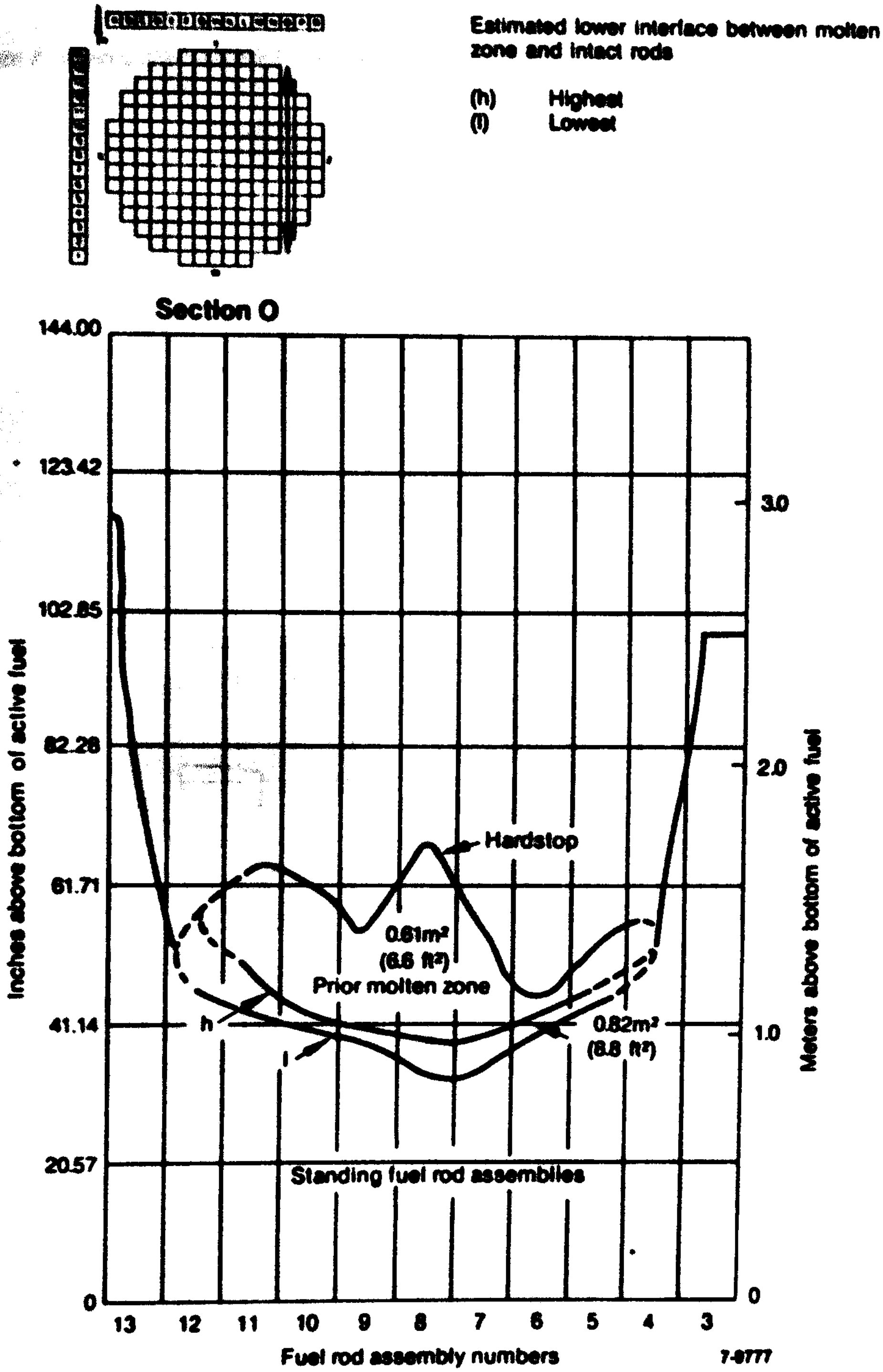
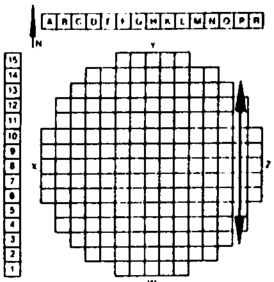


Figure B-13. End-state bounds for intact rods (O-column of fuel assemblies).



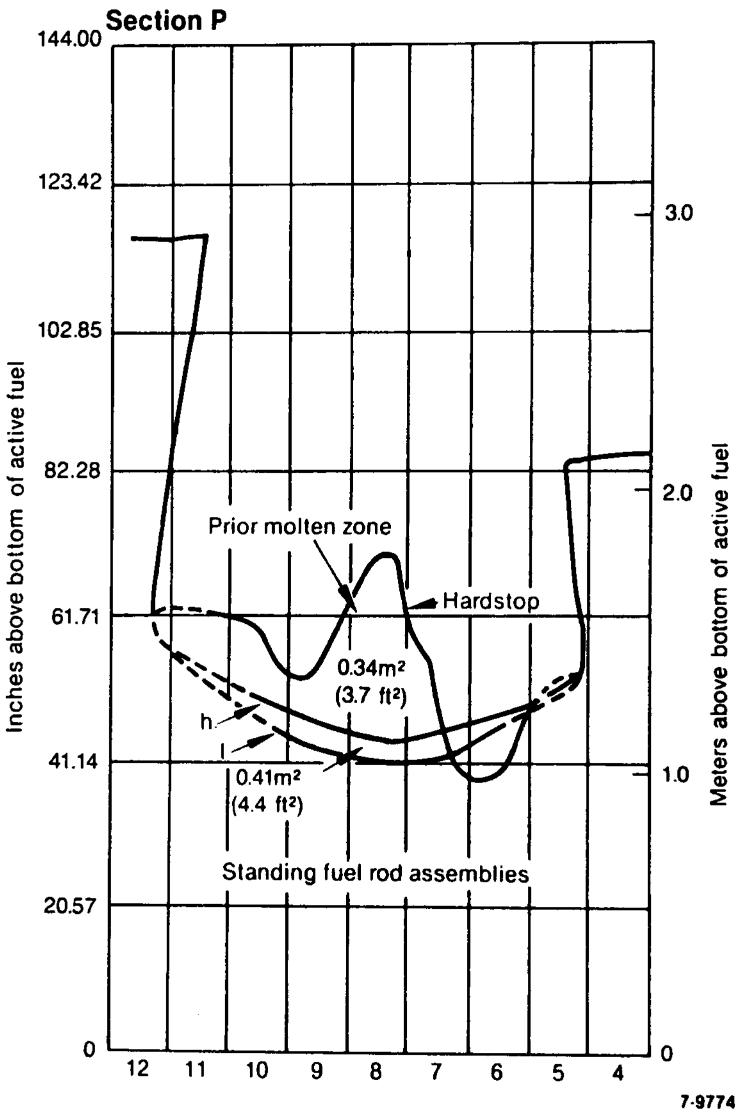


Figure B-14. End-state bounds for intact rods (P-column of fuel assemblies).

TABLE B-1. FUEL ASSEMBLY BURNUP
(MWd/MT)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
A6	966	2136	2632	2617	2522	2290	1416
A7	1031	2260	3053	3380	3321	2970	1984
A8	1334	3050	3738	3794	3717	3410	2134
A9	1032	2260	3054	3381	3321	2971	1984
A10	967	2186	2632	2618	2523	2290	1416
B4	912	1807	2107	2101	2020	1845	1205
B5	1456	2975	3401	3347	3295	3075	1936
B6	1724	3854	4603	4567	4455	4163	2681
B7	1650	3649	4387	4384	4264	3966	2582
B8	1832	4507	5551	5572	5500	5163	3263
B9	1650	3649	4387	4385	4264	3966	2582
B10	1724	3854	4603	4568	4456	4163	2682
B11	1457	2975	3402	3348	3296	3075	1936
B12	912	1807	2108	2102	2021	1845	1205
C3	1003	1973	2220	2170	2131	1996	1279
C4	1416	2741	3142	3122	3040	2837	1895
C5	1589	3219	3630	3478	3361	3194	2125
C6	1848	3928	4342	4016	3906	3856	2591
C7	1872	3801	4370	4264	4106	3840	2548
C8	1974	4101	4832	4794	4611	4261	2811
C9	1872	3801	4371	4264	4106	3840	2548
C10	1914	3864	4303	4063	3926	3811	2612
C11	1590	3220	3630	3479	3362	3194	2125
C12	1417	2742	3143	3124	3041	2838	1895
C13	1003	1973	2220	2171	2131	1997	1279
D2	912	1807	2108	2102	2020	1845	1205
D3	1417	2741	3142	3123	3040	2837	1895
D4	1580	3158	3560	3413	3265	3064	2039
D5	1804	3806	4188	3818	3645	3615	2514
D6	1781	3964	3862	2925	2855	3419	2582
D7	2072	4060	4480	4224	4067	3939	2726
D8	1708	3347	3920	3790	3401	3061	2240
D9	2072	4060	4480	4225	4067	3940	2726
D10	1781	3964	3862	2925	2855	3419	2582
D11	1804	3807	4189	3819	3646	3615	2515
D12	1581	3158	3560	3413	3266	3065	2039
D13	1417	2742	3143	3124	3041	2838	1895
D14	912	1807	2108	2102	2021	1845	1205
E2	1457	2975	3402	3347	3295	3075	1936
E3	1597	3251	3604	3401	3310	3192	2107
E4	1908	3879	4127	3712	3612	3625	2454
E5	1438	3644	4197	3805	3635	3646	2489
E6	1885	4121	4589	4207	4049	4014	2741
E7	2020	3929	4421	4301	4167	3891	2547
E8	2153	4161	4774	4723	4529	4147	2735
E9	1931	3925	4485	4350	4186	3909	2569

TABLE B-1. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
E10	1885	4121	4589	4208	4050	4014	2741
E11	1439	3644	4198	3806	3636	3646	2489
E12	1909	3880	4128	3713	3612	3625	2454
E13	1597	3251	3604	3401	3311	3192	2108
E14	1457	2975	3402	3348	3295	3075	1936
F1	967	2186	2632	2618	2523	2290	1416
F2	1727	3917	4666	4598	4479	4192	2683
F3	1850	3845	4301	4035	3890	3790	2593
F4	1736	3902	3794	2867	2829	3416	2569
F5	2025	4036	4470	4195	4024	3921	2740
F6	2042	4021	4559	4435	4263	3950	2586
F7	2141	4212	4933	4936	4714	4274	2819
F8	1802	3845	4675	4748	4572	4148	2689
F9	2141	4212	4933	4936	4714	4274	2819
F10	2042	4021	4559	4436	4263	3950	2586
F11	2026	4036	4470	4195	4025	3921	2740
F12	1737	3902	3794	2867	2830	3416	2569
F13	1872	3883	4267	3984	3920	3854	2548
F14	1727	3917	4667	4598	4479	4192	2683
F15	967	2186	2633	2618	2523	2290	1416
G1	1032	2260	3053	3380	3321	2970	1984
G2	1699	3587	4276	4324	4252	3949	2552
G3	1868	3838	4395	4265	4133	3907	2594
G4	2060	4078	4494	4208	4036	3925	2728
G5	2076	3915	4393	4294	4147	3878	2606
G6	2184	4237	4921	4952	4804	4384	2847
G7	2053	4089	4836	4904	4739	4270	2735
G8	2105	4284	5236	5421	5228	4678	3042
G9	2053	4089	4836	4904	4739	4270	2734
G10	2184	4237	4921	4952	4805	4385	2847
G11	1939	3888	4439	4314	4145	3875	2578
G12	2060	4077	4494	4208	4036	3925	2728
G13	1868	3838	4395	4265	4133	3907	2593
G14	1699	3587	4276	4324	4252	3949	2552
G15	1032	2260	3053	3381	3321	2970	1984
H1	1334	3050	3738	3794	3717	3409	2134
H2	1832	4507	5550	5572	5500	5163	3262
H3	1974	4100	4831	4793	4611	4260	2811
H4	1708	3346	3919	3789	3400	3060	2240
H5	2152	4160	4773	4722	4528	4146	2735
H6	1801	3844	4674	4747	4572	4147	2688
H7	2104	4283	5235	5421	5228	4678	3042
H8	2057	4799	6009	6213	6117	5571	3464
H9	2104	4283	5235	5421	5228	4678	3042
H10	1801	3844	4674	4747	4572	4147	2688
H11	2152	4160	4773	4722	4528	4146	2735
H12	1707	3346	3919	3789	3400	3060	2240

TABLE B-1. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
H13	1973	4100	4831	4793	4611	4260	2811
H14	1832	4507	5550	5572	5500	5163	3262
H15	1334	3050	3737	3794	3717	3409	2134
K1	1032	2260	3053	3380	3321	2970	1984
K2	1699	3587	4276	4324	4252	3949	2552
K3	1868	3838	4394	4265	4132	3907	2593
K4	2059	4077	4494	4208	4036	3925	2728
K5	2002	3911	4418	4315	4185	3904	2553
K6	2183	4236	4920	4951	4804	4384	2847
K7	2053	4088	4835	4904	4738	4270	2734
K8	2104	4283	5234	5421	5227	4677	3041
K9	2052	4088	4834	4904	4738	4270	2734
K10	2183	4236	4920	4951	4804	4384	2846
K11	1977	3952	4487	4320	4121	3865	2603
K12	2059	4076	4493	4207	4035	3924	2728
K13	1867	3837	4394	4265	4132	3907	2593
K14	1698	3586	4275	4324	4252	3949	2552
K15	1031	2259	3053	3380	3321	2970	1984
L1	967	2186	2632	2618	2523	2290	1416
L2	1726	3916	4666	4597	4478	4191	2683
L3	1789	3792	4212	3927	3839	3796	2565
L4	1735	3901	3792	2866	2828	3416	2568
L5	2025	4034	4468	4193	4023	3920	2739
L6	240	4019	4557	4434	4261	3949	2585
L7	2139	4210	4931	4934	4713	4273	2819
L8	1800	3843	4673	4746	4571	4147	2688
L9	2139	4209	4931	4934	4713	4273	2819
L10	2040	4019	4557	4433	4261	3949	2585
L11	2024	4033	4468	4193	4023	3919	2739
L12	1734	3899	3792	2865	2828	3415	2568
L13	1862	3775	4245	4046	3917	3784	2583
L14	1725	3915	4665	4597	4478	4191	2683
L15	966	2186	2632	2618	2523	2290	1416
M2	1456	2974	3401	3347	3295	3075	1936
M3	1596	3249	3602	3400	3309	3191	2107
M4	1907	3878	4126	3711	3611	3624	2453
M5	1437	3642	4195	3804	3634	3645	2488
M6	1883	4118	4586	4205	4047	4012	2740
M7	1992	3943	4452	4295	4124	3870	2580
M8	2151	4157	4771	4720	4527	4145	2734
M9	1970	3986	4473	4286	4157	3932	2567
M10	1882	4117	4585	4205	4047	4012	2740
M11	1436	3641	4195	3804	3634	3645	2488
M12	1906	3877	4125	3710	3610	3623	2453
M13	1595	3249	3602	3400	3309	3191	2107
M14	1455	2973	3400	3346	3294	3074	1936
N2	911	1806	2107	2101	2020	1844	1204

TABLE B-1. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
N3	1416	2740	3141	3122	3039	2837	1894
N4	1579	3155	3558	3411	3264	3064	2038
N5	1802	3803	4186	3816	3644	3613	2514
N6	1779	3960	3859	2922	2853	3417	2581
N7	2069	4056	4476	4222	4064	3937	2725
N8	1705	3342	3916	3787	3398	3058	2240
N9	2069	4055	4476	4221	4064	3937	2725
N10	1779	3960	3858	2922	2852	3417	2581
N11	1801	3802	4185	3815	3643	3612	2513
N12	1578	3154	3557	3411	3264	3063	2039
N13	1415	2738	3140	3121	3038	2836	1895
N14	911	1805	2106	2101	2020	1844	1204
O3	1002	1971	2218	2170	2130	1996	1278
O4	1415	2739	3141	3121	3039	2836	1894
O5	1587	3216	3627	3476	3359	3192	2124
O6	1787	3805	4279	4010	3872	3788	2589
O7	1859	3797	4367	4261	4103	3838	2547
O8	1971	4096	4828	4790	4609	4258	2810
O9	1869	3796	4367	4261	4103	3838	2547
O10	1752	3756	4237	3982	3864	3796	2593
O11	1587	3216	3626	3476	3359	3192	2124
O12	1414	2738	3139	3120	3038	2836	1894
O13	1002	1971	2218	2169	2130	1995	1278
P4	911	1805	2106	2100	2019	1844	1204
P5	1455	2972	3399	3345	3293	3074	1935
P6	1722	3850	4599	4565	4453	4161	2680
P7	1648	3645	4383	4382	4262	3964	2581
P8	1829	4506	5547	5569	5498	5161	3261
P9	1647	3645	4383	4381	4261	3964	2581
P10	1721	3850	4599	4565	4453	4161	2680
P11	1454	2971	3398	3344	3293	3073	1935
P12	910	1804	2105	2100	2019	1843	1204
R6	965	2183	2630	2616	2521	2288	1416
R7	1029	2256	3050	3378	3319	2969	1983
R8	1332	3046	3734	3791	3715	3408	2133
R9	1029	2256	3050	3378	3319	2969	1983
R10	964	2183	2629	2615	2521	2288	1415

TABLE B-2. FRACTION OF FUEL ASSEMBLY AXIAL REGION CONTAINING INTACT RODS (UPPER BOUND ESTIMATE)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
A6	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A7	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A8	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A9	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A10	1.000	1.000	1.000	1.000	1.000	1.000	1.000
B4	0.000	0.000	1.000	1.000	1.000	1.000	1.000
B5	0.000	0.000	0.000	0.000	0.800	1.000	1.000
B6	0.000	0.000	0.000	0.000	0.800	1.000	1.000
B7	0.000	0.000	0.000	0.000	0.800	1.000	1.000
B8	0.000	0.000	0.000	0.000	0.850	1.000	1.000
B9	0.000	0.000	0.000	0.000	0.900	1.000	1.000
B10	0.000	0.000	0.000	0.000	0.950	1.000	1.000
B11	0.000	0.000	0.400	1.000	1.000	1.000	1.000
B12	0.000	0.000	1.000	1.000	1.000	1.000	1.000
C3	0.000	0.000	0.000	0.700	1.000	1.000	1.000
C4	0.000	0.000	0.000	0.000	0.550	1.000	1.000
C5	0.000	0.000	0.000	0.000	0.450	1.000	1.000
C6	0.000	0.000	0.000	0.000	0.400	1.000	1.000
C7	0.000	0.000	0.000	0.000	0.400	1.000	1.000
C8	0.000	0.000	0.000	0.000	0.450	1.000	1.000
C9	0.000	0.000	0.000	0.000	0.500	1.000	1.000
C10	0.000	0.000	0.000	0.000	0.700	1.000	1.000
C11	0.000	0.000	0.000	0.000	0.800	1.000	1.000
C12	0.000	0.000	0.500	1.000	1.000	1.000	1.000
C13	0.200	1.000	1.000	1.000	1.000	1.000	1.000
D2	0.000	0.600	1.000	1.000	1.000	1.000	1.000
D3	0.000	0.000	0.000	0.000	0.450	1.000	1.000
D4	0.000	0.000	0.000	0.000	0.300	1.000	1.000
D5	0.000	0.000	0.000	0.000	0.150	1.000	1.000
D6	0.000	0.000	0.000	0.000	0.100	1.000	1.000
D7	0.000	0.000	0.000	0.000	0.100	1.000	1.000
D8	0.000	0.000	0.000	0.000	0.150	1.000	1.000
D9	0.000	0.000	0.000	0.000	0.250	1.000	1.000
D10	0.000	0.000	0.000	0.000	0.400	1.000	1.000
D11	0.000	0.000	0.000	0.000	0.700	1.000	1.000
D12	0.000	0.000	0.000	0.000	1.000	1.000	1.000
D13	0.000	0.000	0.000	0.900	1.000	1.000	1.000
D14	0.000	1.000	1.000	1.000	1.000	1.000	1.000
E2	0.000	0.000	0.000	1.000	1.000	1.000	1.000
E3	0.000	0.000	0.000	0.000	0.350	1.000	1.000
E4	0.000	0.000	0.000	0.000	0.150	1.000	1.000
E5	0.000	0.000	0.000	0.000	0.000	0.950	1.000
E6	0.000	0.000	0.000	0.000	0.000	0.850	1.000
E7	0.000	0.000	0.000	0.000	0.000	0.800	1.000
E8	0.000	0.000	0.000	0.000	0.000	0.850	1.000
E9	0.000	0.000	0.000	0.000	0.000	0.950	1.000
E10	0.000	0.000	0.000	0.000	0.150	1.000	1.000
E11	0.000	0.000	0.000	0.000	0.300	1.000	1.000
E12	0.000	0.000	0.000	0.000	0.550	1.000	1.000
E13	0.000	0.000	0.000	0.000	0.900	1.000	1.000

TABLE B-2. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
E14	0.000	0.000	0.450	1.000	1.000	1.000	1.000
F1	0.000	0.400	1.000	1.000	1.000	1.000	1.000
F2	0.000	0.000	0.000	0.000	0.600	1.000	1.000
F3	0.000	0.000	0.000	0.000	0.250	1.000	1.000
F4	0.000	0.000	0.000	0.000	0.000	0.950	1.000
F5	0.000	0.000	0.000	0.000	0.000	0.800	1.000
F6	0.000	0.000	0.000	0.000	0.000	0.700	1.000
F7	0.000	0.000	0.000	0.000	0.000	0.600	1.000
F8	0.000	0.000	0.000	0.000	0.000	0.500	1.000
F9	0.000	0.000	0.000	0.000	0.000	0.650	1.000
F10	0.000	0.000	0.000	0.000	0.000	0.850	1.000
F11	0.000	0.000	0.000	0.000	0.000	1.000	1.000
F12	0.000	0.000	0.000	0.000	0.300	1.000	1.000
F13	0.000	0.000	0.000	0.000	0.600	1.000	1.000
F14	0.000	0.000	0.000	0.700	1.000	1.000	1.000
F15	0.000	0.700	1.000	1.000	1.000	1.000	1.000
G1	0.000	0.000	0.300	1.000	1.000	1.000	1.000
G2	0.000	0.000	0.000	0.000	0.550	1.000	1.000
G3	0.000	0.000	0.000	0.000	0.200	1.000	1.000
G4	0.000	0.000	0.000	0.000	0.050	1.000	1.000
G5	0.000	0.000	0.000	0.000	0.000	0.900	1.000
G6	0.000	0.000	0.000	0.000	0.000	0.650	1.000
G7	0.000	0.000	0.000	0.000	0.000	0.400	1.000
G8	0.000	0.000	0.000	0.000	0.000	0.200	1.000
G9	0.000	0.000	0.000	0.000	0.000	0.400	1.000
G10	0.000	0.000	0.000	0.000	0.000	0.600	1.000
G11	0.000	0.000	0.000	0.000	0.000	0.850	1.000
G12	0.000	0.000	0.000	0.000	0.100	1.000	1.000
G13	0.000	0.000	0.000	0.000	0.400	1.000	1.000
G14	0.000	0.000	0.000	0.000	0.600	1.000	1.000
G15	0.000	0.000	0.300	1.000	1.000	1.000	1.000
H1	0.000	0.000	0.450	1.000	1.000	1.000	1.000
H2	0.000	0.000	0.000	0.000	0.600	1.000	1.000
H3	0.000	0.000	0.000	0.000	0.250	1.000	1.000
H4	0.000	0.000	0.000	0.000	0.150	1.000	1.000
H5	0.000	0.000	0.000	0.000	0.000	1.000	1.000
H6	0.000	0.000	0.000	0.000	0.000	0.800	1.000
H7	0.000	0.000	0.000	0.000	0.000	0.450	1.000
H8	0.000	0.000	0.000	0.000	0.000	0.150	1.000
H9	0.000	0.000	0.000	0.000	0.000	0.200	1.000
H10	0.000	0.000	0.000	0.000	0.000	0.400	1.000
H11	0.000	0.000	0.000	0.000	0.000	0.650	1.000
H12	0.000	0.000	0.000	0.000	0.000	0.950	1.000
H13	0.000	0.000	0.000	0.000	0.250	1.000	1.000
H14	0.000	0.000	0.000	0.000	0.600	1.000	1.000
H15	0.000	0.600	1.000	1.000	1.000	1.000	1.000
K1	0.000	0.000	1.000	1.000	1.000	1.000	1.000
K2	0.000	0.000	0.000	0.000	0.400	1.000	1.000
K3	0.000	0.000	0.000	0.000	0.250	1.000	1.000
K4	0.000	0.000	0.000	0.000	0.150	1.000	1.000
K5	0.000	0.000	0.000	0.000	0.100	1.000	1.000

TABLE B-2. (continued)

Fuel Assembly	(Top) 1	Region 2	3	4	5	6	(Bottom) 7
K6	0.000	0.000	0.000	0.000	0.000	0.900	1.000
K7	0.000	0.000	0.000	0.000	0.000	0.500	1.000
K8	0.000	0.000	0.000	0.000	0.000	0.150	1.000
K9	0.000	0.000	0.000	0.000	0.000	0.000	1.000
K10	0.000	0.000	0.000	0.000	0.000	0.250	1.000
K11	0.000	0.000	0.000	0.000	0.000	0.600	1.000
K12	0.000	0.000	0.000	0.000	0.000	0.900	1.000
K13	0.000	0.000	0.000	0.000	0.250	1.000	1.000
K14	0.000	0.000	0.000	0.000	0.700	1.000	1.000
K15	0.000	0.000	0.150	1.000	1.000	1.000	1.000
L1	0.000	0.000	0.700	1.000	1.000	1.000	1.000
L2	0.000	0.000	0.000	0.000	0.700	1.000	1.000
L3	0.000	0.000	0.000	0.000	0.300	1.000	1.000
L4	0.000	0.000	0.000	0.000	0.200	1.000	1.000
L5	0.000	0.000	0.000	0.000	0.100	1.000	1.000
L6	0.000	0.000	0.000	0.000	0.000	0.900	1.000
L7	0.000	0.000	0.000	0.000	0.000	0.550	1.000
L8	0.000	0.000	0.000	0.000	0.000	0.200	1.000
L9	0.000	0.000	0.000	0.000	0.000	0.150	1.000
L10	0.000	0.000	0.000	0.000	0.000	0.300	1.000
L11	0.000	0.000	0.000	0.000	0.000	0.700	1.000
L12	0.000	0.000	0.000	0.000	0.000	0.950	1.000
L13	0.000	0.000	0.000	0.000	0.300	1.000	1.000
L14	0.000	0.000	0.150	1.000	1.000	1.000	1.000
L15	0.000	0.550	1.000	1.000	1.000	1.000	1.000
M2	0.000	0.000	0.000	0.650	1.000	1.000	1.000
M3	0.000	0.000	0.000	0.000	0.450	1.000	1.000
M4	0.000	0.000	0.000	0.000	0.300	1.000	1.000
M5	0.000	0.000	0.000	0.000	0.150	1.000	1.000
M6	0.000	0.000	0.000	0.000	0.000	0.950	1.000
M7	0.000	0.000	0.000	0.000	0.000	0.700	1.000
M8	0.000	0.000	0.000	0.000	0.000	0.400	1.000
M9	0.000	0.000	0.000	0.000	0.000	0.400	1.000
M10	0.000	0.000	0.000	0.000	0.000	0.700	1.000
M11	0.000	0.000	0.000	0.000	0.000	0.900	1.000
M12	0.000	0.000	0.000	0.000	0.100	1.000	1.000
M13	0.000	0.000	0.000	0.000	0.400	1.000	1.000
M14	0.000	0.000	0.000	0.050	1.000	1.000	1.000
N2	0.000	0.500	1.000	1.000	1.000	1.000	1.000
N3	0.000	0.000	0.000	0.000	0.600	1.000	1.000
N4	0.000	0.000	0.000	0.000	0.350	1.000	1.000
N5	0.000	0.000	0.000	0.000	0.100	1.000	1,000
N6	0.000	0.000	0.000	0.000	0.000	0.900	1.000
N7	0.000	0.000	0.000	0.000	0.000	0.750	1.000
N8	0.000	0.000	0.000	0.000	0.000	0.600	1.000
N9	0.000	0.000	0.000	0.000	0.000	0.700	1.000
N10	0.000	0.000	0.000	0.000	0.000	0.850	1.000
N11	0.000	0.000	0.000	0.000	0.100	1.000	1.000
N12	0.000	0.000	0.000	0.000	0.450	1.000	1.000
N13	0.000	0.000	0.000	0.000	0.750	1.000	1.000
N14	0.000	0.000	0.000	0.800	1.000	1.000	1.000

TABLE B-2. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
03	0.000	0.000	0.500	1.000	1.000	1.000	1.000
04	0.000	0.000	0.000	0.150	1.000	1.000	1.000
05	0.000	0.000	0.000	0.000	0.300	1.000	1.000
06	0.000	0.000	0.000	0.000	0.100	1.000	1.000
07	0.000	0.000	0.000	0.000	0.000	1.000	1.000
08	0.000	0.000	0.000	0.000	0.000	0.950	1.000
09	0.000	0.000	0.000	0.000	0.000	1.000	1.000
010	0.000	0.000	0.000	0.000	0.150	1.000	1.000
011	0.000	0.000	0.000	0.000	0.400	1.000	1.000
012	0.000	0.000	0.000	0.000	0.800	1.000	1.000
013	0.000	0.000	0.000	0.900	1.000	1.000	1.000
P4	0.000	0.000	0.100	1.000	1.000	1.000	1.000
P5	0.000	0.000	0.000	0.000	0.700	1.000	1.000
P6	0.000	0.000	0.000	0.000	0.400	1.000	1.000
P7	0.000	0.000	0.000	0.000	0.250	1.000	1.000
P8	0.000	0.000	0.000	0.000	0.250	1.000	1.000
P9	0.000	0.000	0.000	0.000	0.300	1.000	1.000
P10	0.000	0.000	0.000	0.000	0.500	1.000	1.000
P11	0.000	0.000	0.000	0.000	0.700	1.000	1.000
P12	0.000	0.400	1.000	1.000	1.000	1.000	1.000
R6	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R7	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R8	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R9	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R10	0.000	0.000	0.000	1.000	1.000	1.000	1.000

TABLE B-3. FRACTION OF FUEL ASSEMBLY AXIAL REGION CONTAINING INTACT RODS (LOWER BOUND ESTIMATE)

Fuel Assembly	Region (Top) 1	2	3	4	5	6	(Bottom) 7
A6	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A7	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A8	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A9	1.000	1.000	1.000	1.000	1.000	1.000	1.000
A10	1.000	1.000	1.000	1.000	1.000	1.000	1.000
B4	0.000	0.000	0.000	1.000	1.000	1.000	1.000
B5	0.000	0.000	0.000	0.000	0.700	1.000	1.000
B6	0.000	0.000	0.000	0.000	0.600	1.000	1.000
B7	0.000	0.000	0.000	0.000	0.650	1.000	1.000
B8	0.000	0.000	0.000	0.000	0.700	1.000	1.000
B9	0.000	0.000	0.000	0.000	0.800	1.000	1.000
B10	0.000	0.000	0.000	0.000	0.950	1.000	1.000
B11	0.000	0.000	0.400	1.000	1.000	1.000	1.000
B12	0.000	0.000	1.000	1.000	1.000	1.000	1.000
C3	0.000	0.000	0.000	0.000	0.600	1.000	1.000
C4	0.000	0.000	0.000	0.000	0.400	1.000	1.000
C5	0.000	0.000	0.000	0.000	0.350	1.000	1.000
C6	0.000	0.000	0.000	0.000	0.300	1.000	1.000
C7	0.000	0.000	0.000	0.000	0.300	1.000	1.000
C8	0.000	0.000	0.000	0.000	0.350	1.000	1.000
C9	0.000	0.000	0.000	0.000	0.450	1.000	1.000
C10	0.000	0.000	0.000	0.000	0.600	1.000	1.000
C11	0.000	0.000	0.000	0.000	0.800	1.000	1.000
C12	0.000	0.000	0.200	1.000	1.000	1.000	1.000
C13	0.000	0.400	1.000	1.000	1.000	1.000	1.000
D2	0.000	0.000	0.500	1.000	1.000	1.000	1.000
D3	0.000	0.000	0.000	0.000	0.300	1.000	1.000
D4	0.000	0.000	0.000	0.000	0.200	1.000	1.000
D5	0.000	0.000	0.000	0.000	0.050	1.000	1.000
D6	0.000	0.000	0.000	0.000	0.000	1.000	1.000
D7	0.000	0.000	0.000	0.000	0.000	1.000	1.000
D8	0.000	0.000	0.000	0.000	0.000	1.000	1.000
D9	0.000	0.000	0.000	0.000	0.100	1.000	1.000
D10	0.000	0.000	0.000	0.000	0.250	1.000	1.000
D11	0.000	0.000	0.000	0.000	0.500	1.000	1.000
D12	0.000	0.000	0.000	0.000	0.800	1.000	1.000
D13	0.000	0.000	0.000	0.500	1.000	1.000	1.000
D14	0.000	0.000	1.000	1.000	1.000	1.000	1.000
E2	0.000	0.000	0.000	0.250	1.000	1.000	1.000
E3	0.000	0.000	0.000	0.000	0.200	1.000	1.000
E4	0.000	0.000	0.000	0.000	0.000	0.900	1.000
E5	0.000	0.000	0.000	0.000	0.000	0.800	1.000
E6	0.000	0.000	0.000	0.000	0.000	0.700	1.000
E7	0.000	0.000	0.000	0.000	0.000	0.700	1.000
E8	0.000	0.000	0.000	0.000	0.000	0.750	1.000
E9	0.000	0.000	0.000	0.000	0.000	0.800	1.000
E10	0.000	0.000	0.000	0.000	0.000	0.900	1.000
E11	0.000	0.000	0.000	0.000	0.100	1.000	1.000
E12	0.000	0.000	0.000	0.000	0.300	1.000	1.000
E13	0.000	0.000	0.000	0.000	0.750	1.000	1.000

TABLE B-3. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
E14	0.000	0.000	0.000	0.700	1.000	1.000	1.000
F1	0.000	0.000	0.700	1.000	1.000	1.000	1.000
F2	0.000	0.000	0.000	0.000	0.400	1.000	1.000
F3	0.000	0.000	0.000	0.000	0.050	1.000	1.000
F4	0.000	0.000	0.000	0.000	0.000	0.800	1.000
F5	0.000	0.000	0.000	0.000	0.000	0.700	1.000
F6	0.000	0.000	0.000	0.000	0.000	0.450	1.000
F7	0.000	0.000	0.000	0.000	0.000	0.400	1.000
F8	0.000	0.000	0.000	0.000	0.000	0.400	1.000
F9	0.000	0.000	0.000	0.000	0.000	0.450	1.000
F10	0.000	0.000	0.000	0.000	0.000	0.600	1.000
F11	0.000	0.000	0.000	0.000	0.000	0.800	1.000
F12	0.000	0.000	0.000	0.000	0.000	1.000	1.000
F13	0.000	0.000	0.000	0.000	0.400	1.000	1.000
F14	0.000	0.000	0.000	0.400	1.000	1.000	1.000
F15	0.000	0.000	0.950	1.000	1.000	1.000	1.000
G1	0.000	0.000	0.000	0.500	1.000	1.000	1.000
G2	0.000	0.000	0.000	0.000	0.350	1.000	1.000
G3	0.000	0.000	0.000	0.000	0.050	1.000	1.000
G4	0.000	0.000	0.000	0.000	0.000	0.800	1.000
G5	0.000	0.000	0.000	0.000	0.000	0.500	1.000
G6	0.000	0.000	0.000	0.000	0.000	0.300	1.000
G7	0.000	0.000	0.000	0.000	0.000	0.200	1.000
G8	0.000	0.000	0.000	0.000	0.000	0.150	1.000
G9	0.000	0.000	0.000	0.000	0.000	0.200	1.000
G10	0.000	0.000	0.000	0.000	0.000	0.350	1.000
G11	0.000	0.000	0.000	0.000	0.000	0.600	1.000
G12	0.000	0.000	0.000	0.000	0.000	0.850	1.000
G13	0.000	0.000	0.000	0.000	0.150	1.000	1.000
G14	0.000	0.000	0.000	0.000	0.450	1.000	1.000
G15	0.000	0.000	0.000	0.500	1.000	1.000	1.000
H1	0.000	0.000	0.000	0.750	1.000	1.000	1.000
H2	0.000	0.000	0.000	0.000	0.300	1.000	1.000
H3	0.000	0.000	0.000	0.000	0.000	1.000	1.000
H4	0.000	0.000	0.000	0.000	0.000	1.000	1.000
H5	0.000	0.000	0.000	0.000	0.000	0.750	1.000
H6	0.000	0.000	0.000	0.000	0.000	0.400	1.000
H7	0.000	0.000	0.000	0.000	0.000	0.150	1.000
H8	0.000	0.000	0.000	0.000	0.000	0.000	0.900
H9	0.000	0.000	0.000	0.000	0.000	0.000	0.750
H10	0.000	0.000	0.000	0.000	0.000	0.000	0.800
H11	0.000	0.000	0.000	0.000	0.000	0.000	1.000
H12	0.000	0.000	0.000	0.000	0.000	0.300	1.000
H13	0.000	0.000	0.000	0.000	0.000	0.600	1.000
H14	0.000	0.000	0.000	0.000	0.000	0.950	1.000
H15	0.000	0.000	0.000	0.000	0.350	1.000	1.000
K1	0.000	0.000	0.700	1.000	1.000	1.000	1.000
K2	0.000	0.000	0.300	1.000	1.000	1.000	1.000
K3	0.000	0.000	0.000	0.000	0.300	1.000	1.000
K4	0.000	0.000	0.000	0.000	0.000	1.000	1.000
K5	0.000	0.000	0.000	0.000	0.000	0.700	1.000
					0.000	0.400	1.000

TABLE B-3. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
K6	0.000	0.000	0.000	0.000	0.000	0.050	1.000
K7	0.000	0.000	0.000	0.000	0.000	0.000	0.750
K8	0.000	0.000	0.000	0.000	0.000	0.000	0.400
K9	0.000	0.000	0.000	0.000	0.000	0.000	0.300
K10	0.000	0.000	0.000	0.000	0.000	0.000	0.900
K11	0.000	0.000	0.000	0.000	0.000	0.250	1.000
K12	0.000	0.000	0.000	0.000	0.000	0.650	1.000
K13	0.000	0.000	0.000	0.000	0.000	0.950	1.000
K14	0.000	0.000	0.000	0.000	0.350	1.000	1.000
K15	0.000	0.000	0.000	0.250	1.000	1.000	1.000
L1	0.000	0.000	0.000	0.700	1.000	1.000	1.000
L2	0.000	0.000	0.000	0.000	0.400	1.000	1.000
L3	0.000	0.000	0.000	0.000	0.100	1.000	1.000
L4	0.000	0.000	0.000	0.000	0.000	0.800	1.000
L5	0.000	0.000	0.000	0.000	0.000	0.450	1.000
L6	0.000	0.000	0.000	0.000	0.000	0.100	1.000
L7	0.000	0.000	0.000	0.000	0.000	0.000	0.800
L8	0.000	0.000	0.000	0.000	0.000	0.000	0.600
L9	0.000	0.000	0.000	0.000	0.000	0.000	0.700
L10	0.000	0.000	0.000	0.000	0.000	0.000	1.000
L11	0.000	0.000	0.000	0.000	0.000	0.300	1.000
L12	0.000	0.000	0.000	0.000	0.000	0.700	1.000
L13	0.000	0.000	0.000	0.000	0.000	1.000	1.000
L14	0.000	0.000	0.000	0.000	0.400	1.000	1.000
L15	0.000	0.000	0.500	1.000	1.000	1.000	1.000
M2	0.000	0.000	0.000	0.000	1.000	1.000	1.000
M3	0.000	0.000	0.000	0.000	0.250	1.000	1.000
M4	0.000	0.000	0.000	0.000	0.000	1.000	1.000
M5	0.000	0.000	0.000	0.000	0.000	0.700	1.000
M6	0.000	0.000	0.000	0.000	0.000	0.300	1.000
M7	0.000	0.000	0.000	0.000	0.000	0.050	1.000
M8	0.000	0.000	0.000	0.000	0.000	0.000	1.000
M9	0.000	0.000	0.000	0.000	0.000	0.150	1.000
M10	0.000	0.000	0.000	0.000	0.000	0.350	1.000
M11	0.000	0.000	0.000	0.000	0.000	0.550	1.000
M12	0.000	0.000	0.000	0.000	0.000	0.850	1.000
M13	0.000	0.000	0.000	0.000	0.150	1.000	1.000
M14	0.000	0.000	0.000	0.000	0.450	1.000	1.000
N2	0.000	0.000	0.600	1.000	1.000	1.000	1.000
N3	0.000	0.000	0.000	0.000	0.450	1.000	1.000
N4	0.000	0.000	0.000	0.000	0.200	1.000	1.000
N5	0.000	0.000	0.000	0.000	0.000	0.950	1.000
N6	0.000	0.000	0.000	0.000	0.000	0.700	1.000
N7	0.000	0.000	0.000	0.000	0.000	0.400	1.000
N8	0.000	0.000	0.000	0.000	0.000	0.400	1.000
N9	0.000	0.000	0.000	0.000	0.000	0.500	1.000
N10	0.000	0.000	0.000	0.000	0.000	0.600	1.000
N11	0.000	0.000	0.000	0.000	0.000	0.800	1.000
N12	0.000	0.000	0.000	0.000	0.000	1.000	1.000
N13	0.000	0.000	0.000	0.000	0.300	1.000	1.000
N14	0.000	0.000	0.000	0.000	0.800	1.000	1.000

TABLE B-3. (continued)

Fuel Assembly	(Top) 1	2	3	Region 4	5	6	(Bottom) 7
03	0.000	0.000	0.000	1.000	1.000	1.000	1.000
04	0.000	0.000	0.000	0.000	0.750	1.000	1.000
05	0.000	0.000	0.000	0.000	0.200	1.000	1.000
06	0.000	0.000	0.000	0.000	0.000	1.000	1.000
07	0.000	0.000	0.000	0.000	0.000	0.800	1.000
08	0.000	0.000	0.000	0.000	0.000	0.750	1.000
09	0.000	0.000	0.000	0.000	0.000	0.900	1.000
010	0.000	0.000	0.000	0.000	0.000	1.000	1.000
011	0.000	0.000	0.000	0.000	0.150	1.000	1.000
012	0.000	0.000	0.000	0.000	0.400	1.000	1.000
013	0.000	0.000	0.000	0.200	1.000	1.000	1.000
P4	0.000	0.000	0.000	0.750	1.000	1.000	1.000
P5	0.000	0.000	0.000	0.000	0.500	1.000	1.000
P6	0.000	0.000	0.000	0.000	0.000	1.000	1.000
P7	0.000	0.000	0.000	0.000	0.000	1.000	1.000
P8	0.000	0.000	0.000	0.000	0.000	1.000	1.000
P9	0.000	0.000	0.000	0.000	0.150	1.000	1.000
P10	0.000	0.000	0.000	0.000	0.350	1.000	1.000
P11	0.000	0.000	0.000	0.000	0.600	1.000	1.000
P12	0.000	0.000	0.900	1.000	1.000	1.000	1.000
R6	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R7	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R8	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R9	0.000	0.000	0.000	1.000	1.000	1.000	1.000
R10	0.000	0.000	0.000	1.000	1.000	1.000	1.000

TABLE B-4. FRACTIONAL CORE BURNUP WITHIN INTACT ROD REGION

Axial Region	Upper Bound Case Total Burnup	Upper Bound Case Intact Rod Burnup	Upper Bound Case Fraction of Burnup in Intact Rods	Lower Bound Case Total Burnup	Lower Bound Case Intact Rod Burnup	Lower Bound Case Fraction of Burnup in Intact Rods
1	294360	5530	0.019	294360	5330	0.018
2	616777	23817	0.039	616777	12681	0.020
3	709840	56955	0.080	709840	36938	0.052
4	688775	113955	0.16	688775	87510	0.127
5	666330	248371	0.37	666330	187535	0.281
6	630134	548990	0.87	630134	470780	0.747
7	418710	418710	1.0	418710	409740	0.978

Fraction of total core burnup in intact rods (upper bound) = 0.35

Fraction of total core burnup in intact rods (lower bound) = 0.30